Transcript of Proceedings

UNITED STATES OF AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT
THREE MILE ISLAND

DEPOSITION OF: ROGER J. MATTSON

Bethesda, Maryland

August 6, 1979

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

August 13, 1979

TO: Roger Mattson

FROM: Richard S. Mallory, OGC RSM

Enclosed is a copy of the transcript of your deposition before the President's Commission on the Accident at Three Mile Island.

Please read through the transcript carefully and correct any errors (other than unimportant punctuation errors) in black pen on this copy. Correct any errors you can identify in the questions, as well as in your answers. This copy will not be retyped, but will be reproduced as you have marked it, so your corrections should be dark and legible. If you cross out words in the transcript, draw only a single line through them, so that they can still be easily read when the transcript is copied; do not obliterate them.

After you have corrected the transcript, please sign and date the certificate at the end, and type your name under your signature.

You may wish to make a copy of the transcript for yourself before returning the original to me. When you return the transcript, please indicate if you object to making your transcript available to the Commission or to the Commission's investigation of Three Mile Island. Because of Commissioner interest, we would appreciate receiving your corrected copy by c.o.b. Tuesday, August 14, if possible. no object

The President's Commission on Three Mile Island will also be sending you a copy of your transcript with a request to make an "errata sheet" and sign a signature page. Please make up an "errata sheet" based on the copy of the transcript that you have retained and return the errata sheet and signature page to the President's Commission as requested in their letter.

If you have any questions or problems, do not hesitate to call me or the attorney who represented you at the deposition.

Enclosure: Transcript

CERTIFICATE

I certify that I have read this transcript and corrected any errors in the transcription that I have been able to identify, except for unimportant punctuation errors.

Date: 8/16/79 Roger J Mattson

acme

UNITED STATES OF AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND

DEPOSITION OF: ROGER J. MATTSON

Room 1102
New Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland

August 6, 1979
9:30 o'clock a.m.

APPEARANCES:

On Behalf of the Commission:

KEVIN P. KANE, ESQ.
Associate Chief Counsel
DWIGHT H. REILLY
Technical Staff
2100 M Street, N.W.
Washington, D.C. 20037

On Behalf of NRC:

MARK CHOPKO, ESQ.
Office of General Counsel
1717 H Street, N.W.
Washington, D.C.

I N D E X

E X H I B I T S

P R O C E E D I N G S

Whereupon,

ROGER JOSEPH MATTSON

having been duly sworn, was called as a witness herein, and testified as follows:

DIRECT EXAMINATION

BY MR. KANE:

Q Would you state your full name for the record, plese?

A Roger Joseph Mattson.

Q Have you ever had your deposition taken before, Mr. Mattson?

A No.

MR. KANE: Let me briefly explain what we are doing here today. You have been sworn, and although we are sitting in the relative informality of your offices, you should be aware that the testimony you are giving here today has the same force and solemnity as if you were testifying in a court of law.

My questions and your answers are being taken down by the court reporter here. They will be later reduced to a booklet form. You will be presented with a copy of that booklet and given an opportunity to make any corrections you deem necessary.

On the other hand, to the extent that you make corrections in matters which we deem to be significant, we may comment upon those changes and those comments may be adverse to your credibility.

For that reason, it is important to avoid the necessity for changes as much as possible by being as accurate and as precise as we can be now.

In that connection then, I would ask if at any point in this deposition if you don't understand the question, if you feel something needs clarification or elaboration, please feel free to stop to indicate that and we will make the elaboration or clarification on the record at that time.

Last, let me advise you of the two basic groundrules in a deposition. First is that you respond audibly to my questions since the reporter cannot take down a nod of the head or a gesture and secondly, please allow me to finish my questions before you respond even if you know what the question is going to be because the reporter cannot take us both down at the sametime.

Do you have any questions about that?

THE WITNESS: No. I understand.

BY MR. KANE:

Q Fine. Have you brought with you here today a resume which briefly states your educational and employment background, Mr. Mattson?

A Yes.

Q I have here a document entitled "Professional Qualifications, Roger J. Mattson."

Is that that resume?

A Yes, it is.

Q Does this accurately reflect your educational and employment background to date?

A Yes.

MR. KANE: Let's have this marked as Exhibit 1 to the deposition.

(Mattson Exhibit No 1 was marked for identification.)

BY MR. KANE:

Q Mr. Mattson, you have been Director of the Division of Systems Safety in the NRC since July of 1977. Would you generally describe the duties of the Division of Systems Safety and your duties as Director of that Division?

A The Division of Systems Safety performs most of the engineering and technical review of construction permit applications and operating license applications for nuclear power plants.

It is organized into ten technical branches for the conduct of that review. In addition, the Division is responsible for a number of NRC's unresolved safety issues, for ~~or~~ research coordination, for standards coordination and for the analytical support of nuclear power plants/ safety reviews by ~~performance for~~ the Division of Operating Reactors, in addition to the Division of Systems Safety.

Besides the analytical support for the Division of Operating Reactors, we also provide other technical support upon request by the Director of that Division.

There are other day-to-day responsibilities, but those are the major areas as Director of the Division that I am responsible for--managing the resources, maintaining the planning and program coordination for those activities, and as a technical decision maker in appeal of decisions that are not reached below me, probably I need to fill in some words there, by and large, the day-to-day technical decisions are delegated within the Division to assistant directors and branch chiefs who use the Standard Review Plan and regulatory guides in the performance of their review of CP and OL applications.

As new problems arise, decisions on the technical solutions for those new problems are generally carried all the way through to the office director level if they do not require rule making and through to the Commission if they do require rule making, so I am involved as a technical decision maker in part of the line organization chain.

On incidental things that arise in the interpretation of the standing guidelines and documents, oftentimes licensees will appeal staff positions as being too stringent. Those appeals are brought up the line and if not resolved at the assistant director level, then I sit on the appeal and make a technical decision on the matter.

Q In the event that the technical person within the Division of Safety Systems comes across a potentially generic safety problem dealing with reactor systems, would that be passed up through the chain to you?

A Yes. As a matter of fact, before he can do anything generic about its solution, it is required that the top management of the office approve the method that the staff intends to use to solve this new problem, so it would come to me. I would transmit it to the Regulatory Requirements Review Committee if the solution were known.

If the solution were not known, and it wasn't ready for top management to approve, then it would be identified as a generic issue for consideration by the technical activities steering committee and identification as such and assignment of resources and what have you.

Q Would it be fair to say that insofar as there is one location within the NRC where there is a concentration of technical expertise, it would be the

Division of Systems Safety?

A Yes. That is fair. We have no project managers in the Division, so the people, the professionals here are all technical specialists or systems specialists. They have counterparts in the Division of Operating Reactors, but they also have a project management function in the Division of Operating Reactors, so the concentration of technical resources is not as high.

Q Okay. Who has overall responsibility within the Division of Systems Safety for systems integration or systems engineering for a given project?

A Well, there is no single individual anywhere in the Office of Nuclear Reactor Regulation who has responsibility for overall integration of systems engineering. I take that in the technical sense, in the sense that an architect/engineer like Bechtel would have an overall systems engineer on a project usually a senior systems specialist who performs that integrating function.

We perform that integration through the Standard Review Plan or at least that has been the intent since the Standard Review Plan was issued in 1975, and in addition, there is an integration that occurs at a subsystem level, that is, major portions of the nuclear power plant are integrated in the Reactor Systems Branch, the Containment Systems Branch, the Power Systems Branch, the Auxiliary

Systems Branch, and the Instrumentation and Control Systems Branch, so if you take those titles you can see the major building blocks of systems integration.

The ~~INC~~ I and C Branch, for example, integrates instrumentation and controls as a subsystem of the overall nuclear power plant.

The Reactor Systems people integrate generally the nuclear steam supply system, scope of supply, while the Auxiliary Systems Branch and the Power Systems Branch would be the integrators of the balance of plant.

Q But there is no single office within the NRC that does system integration as such, in other words, would occupy the regulatory equivalent of say Bechtel or someone like that, is that correct?

A No. I think you misunderstand what I am saying. Bechtel also organizes itself into these subelements, not precisely the same ones that I have described for NRR, but it is usual in an architect/engineering firm to have a systems integration ~~by either~~ across the entire scope of supply of that architect/engineer, by either a senior individual or a small group.

That counterpart does not exist in NRR, as I understand it; although I wasn't part of the decision-making process, I have been part of explaining it before. As I understand it, the decision to not do that was made when the Standard Review Plan was issued. That is, the Standard Review Plan provided that integration of the various systems elements.

Now there has been an unresolved safety issue identified for two or three years called systems interaction that the ACRS identified. It has been under study and work for the last couple of years. It ~~was~~ has ~~represented as~~ representatives from the Division of Systems Safety and all of the systems branches that I advise, and that is one of the things that is being considered in that unresolved safety issue is how does the integration of systems engineering effects occur ~~most~~ in the design process ~~than~~ and in the review process, ~~but~~ it is possible that out of that problem, the solution of that problem, would come some changes in the way NRR is organized.

Q What is the unresolved safety issue in systems interaction?

A The unresolved issue is to what extent are failure modes and effects in one system considered for their possible interaction, deleterious interaction, with other systems?

For example, a classic I guess is could the air conditioning system failure cause equipment in an ancilliary system to fail, then causing a safety system to fail in ways not considered in the design of the safety system because the systems I just described as an example were handled as discrete packages by three discrete engineering organizations.

Q And that very scenario would lie outside the normal mode of analysis used in SPR which is single failure analysis?

A Yes, and clearly outside the scope of NRC's review of nuclear power plants heretofore.

Q It has not been felt within the NRC that that issue could be resolved by the use of the Bechtel-type of analogy of a single person or a single group of persons whose overall responsibility is system interaction?

A No. I don't think we have dismissed any solution. Through the last couple of years, we have paid higher and higher attention to these system interactions effects. That is, what are the possible intereactions that could have been overlooked in previous designs. ~~and~~ the way we have chosen to do it, lacking proof of a better way, is to expand the cognizance of the system branches, that I described earlier, of the possible deleterious effects of supporting systems on a system that they normally review and effects of this sort have been found. The one I just described is an example.

Q Sure. I take it from the example that you just gave that, of course, the whole systems interaction question extends much beyond interaction of safety-related systems. It would relate to all of the systems in the plant really.

A Yes, in ways that Three Mile Island shows.

Q I wanted to come to that because we did have a discussion during the interview we had prior to this deposition about the Standard Review Plan and how it interacted or did not interact with the design and approval of Three Mile Island 2.

I specifically asked you at that time I think about the containment isolation criteria and the fact that it is my understanding you had containment isolation actuation based upon a single factor PSI in the containment building.

A High pressure.

Q And that the requirement under the Standard Review Plan for diverse actuation of containment isolation on any two out of three factors--I think we discussed at that time how it came about that Three Mile Island 2 simply did not wind up having that requirement imposed but I would like to get that now in the record if I could.

How did it come about that Three Mila Island 2 did not have diverse containment isolation actuation as required under the Standard Review Plan?

A Three Mile Island 2 was not reviewed according to the Standard Review Plan. It was not required to demonstrate conformance with the Standard Review Plan. That may or may not explain why it did not have diverse containment isolation actuation.

There may in fact be plants like Three Mile Island 2 who obtained their CP's or OL's prior to the fall of 1975 when the Standard Review Plan was issued that still have diverse containment isolation actuation.

There it would have been a matter of a particular designer or a particular reviewer insisting that there be diverse actuation. That is, there was a growing or changing state of technology frozen by the Standard Review Plan by 1975 where in some cases there was diverse actuation, in others there wasn't. I can't state for certain how it came to be that Three Mile Island was a plant that did not have diverse containment isolation.

Q The deadline date that you mentioned in 1975 was September did you say?

A I can look it up, the Standard Review Plan I think states when it was first issued--September, 1975.

Q Okay. If I understand what you just said, the approach that was taken was that any project which received a construction permit prior to September, 1975 was not subject to the requirements of the Standard Review Plan?

A That's right.

Q Now Three Mile Island did not obtain its operating license until February of 1978. Why was it felt that a Standard Review Plan which was effective as of September, 1975 should not apply to Three Mile Island 2 or other plants like it because their construction permit was obtained before September, 1975.

A Well, I can give you my understanding of the philosophy that was applied. I cannot give you a firsthand explanation of the logic of the decision makers who made that decision. I was not employed in this office at the time that decision was made, and I only have available hearsay from others and a review of the records that are available from that time.

Q At that time I guess in 1975 you were Director of the Division of Siting Health and Safeguards Standards, Office of Standards Development?

A That's right. As I understand the philosophy, it was that the Standard Review Plan was a codification of existing review requirements with some few exceptions. Those few exceptions ~~involving~~ involved what we call ratchets in the licensing process that were acknowledged to have been put in the Standard Review Plan that went beyond then existing review practice,but were thought by the drafters of the Standard Review Plan and the people then in positions of responsibility in the Office of Nuclear

Reactor Regulation to be justifiable ratchets. so with those few exceptions, the philosophy was that the SRP ~~is~~ was a statement of current practice, and that as a statement of current practice, the plants that had been reviewed at about the time of its generation, that is, in the period say of 1973, 1974, 1975 when the Standard Review Plan was being drafted, would have had approximately the same requirements applied to them case by case by the then existing technical review staff.

Hence it was not necessary to conduct a systematic reappraisal of those plants according to this document.

Now what we came to find in the years after 1975 was that there were significant differences, that there had been significant variability plant to plant, case by case, in the licensing review prior to the issuance of the Standard Review Plan.

That has an element of good news and bad news. The good news is that the Standard Review Plan had done what it was supposed to do. It had achieved some uniformity and consistency in the licensing requirements, and the bad news, of course, was that there were plants that had system designs that weren't as good as some other plants.

Now shortly after the Standard Review Plan was issued in 1975, there are three memos of interest in its implementation. I previously supplied these to the Commission staff at your request after my interview. I don't remember ~~their~~ the dates of the office letters from NRR and I don't recall the specific numbers, but ~~they are~~ there is a supplementary memorandum to one of those office letters.

Generally they say that plants, they start out by first saying that all plants will be reviewed according to the Standard Review Plan, whether they are OL's or CP's, and then they subsequently say that a class of facilities will be grandfathered from the Standard Review Plan, that is, grandfathered from a requirement for the license applicant to demonstrate conformance with the Standard Review Plan.

Instead, the license review staff, the technical staff of the NRC, is required to identify any deviations in a license application relative to the Standard Review Plan, obtain from the licensee justification for those deviations, and then document the rationale for allowing or not allowing the deviations in the safety evaluation report.

Now there were two sets ~~of~~ or stages of grandfathering. One stage required no documentation of deviations. The second stage required staff identification and rationalization of the deviations, and then finally there really is a third stage which is a non-grandfathered plant, those plants in which the applicant must identify any deviations relative to the Standard Review Plan.

Three Mile Island 2 falls in the class of plants which had no identification of deviations relative to the Standard Review Plan.

Q What was the rationale behind committing certain projects to the, if you will, entirely grandfathered under the SRP without even providing any documentation for deviations?

A Well, the rationale must have been their advanced state of construction and licensing review.

Q Why wouldn't it be deemed--let me ask this. Were the provisions of the SRP, the Standard Review Plan, based upon a concensus within the NRC as to what the desirable safety features of a nuclear power plant should be?

A I am not sure what the word concensus means to you. Let me explain how they were written. To me, I think concensus is right, but it may mean something different to you.

The review plans were integrated in a draft form by each of the review branches to describe what they customarily did to a construction permit and an operating license application--what they reviewed, what criteria they applied to it, and what findings they reached, and then a listing of resources or references of an historical nature that were associated with that review.

They were then subject to management review within what was then the Directorate of Technical Review in the Office of Regulation. The director of that directorate at that time was the present Chairman of the NRC, Joe Hendrie.

After the line management review of the input written by the individual branches, and it wasn't a one-way street, there was a lot of iteration back and forth over a period of more than a year; then Dr. Hendrie left NRC and returned to Brookhaven National Laboratory under contract ~~to~~ by Brookhaven, to the then AEC, he completed the final editing and systematic review of the Standard Review Plan from Brookhaven, submitted it to the regulatory staff of the AEC in 1975, so by then it had become the NRC, for promulgation.

It was approved and promulgated by the Director of NRR who would have been Ben Rusche, in September, 1975.

Now that process says that the technical experts in each of the disciplines had input to and iterated upon the contents of the Standard Review Plan.

Q Is it fair to say then, Mr. Mattson, that the Standard Review Plan reflects the official position of the

NRC as to those safety features which should be included in the design of a nuclear power plant?

A Yes, if you understand that the staff has for sometime operated under a philosophy that you may want more for a new plant than what you think is required for an old plant.

Q Yes.

A That is, the Commission's generally worded regulations allow some flexibility in interpretation, and the staff has used that flexibility down through the years to require more of new plants than it has been willing to accept as providing a minimal meeting of the Commission's regulations for old plants.

Q That is the philosophy I wanted to try to focus on, Mr. Mattson. It seems to me what you are saying is that over the years, the operating experience with plants and the development of technology has brought to the attention of the NRC certain aspects of plant design which are safer than other aspects which can be made more acceptable from the point of view of safety, and that eventually those have found their way into the Standard Review Plan and reflect the current thinking on that matter, but that for older plants, the determination was made that for one reason or another, they would not be called upon to comply with at least some of those safety provisions.

You were asked about this when you testified in front of the Presidential Commission on June 1st, 1979. Specifically you were asked about backfitting the Standard Review Plan to older plants, which is I take it what we are talking about.

You said at that time that the decision to not backfit the Standard Review Plan was a conscience decision on the part of the Nuclear Regulatory Commission that was made sometime in 1975.

What did you mean by a conscience decision?

A What I had in mind when I was testifying were the letters of Mr. Rusche that I have previously supplied to the Commission. There was office letters that I was describing here a few minutes ago which ~~consciously~~ chronicled the conscious decisions ~~as~~ to first apply it, and ~~the~~ then not to apply it during 1975 and 1976.

Q What kind of factors entered into determining whether or not the Standard Review Plan should be applied to a plant like TMI 2, for example?

A I think I have just described them. I will summarize them succinctly. One, the philosophy that it was a codification of existing review practice. If you accept that judgment, then it is not necessary to backfit it.

Q Before you go on, I am not sure I understand the point you made there. It is a codification of existing review practice. I understand that. The existing review practice presumably is based upon safety considerations, is that right?

A That's right.

Q And the NRC is not reviewing the plant for any other reason?

A That's right.

Q It is based on safety considerations. Now the determination is made that that is a current practice and that I take it by the way you say that you are inferring that, therefore, there is no reason to apply it in a backfitting sort of way to older plants, and my question is why not?

A I think we are having some difficulty in communicating. I was speaking to the backfit of plants under construction, and I think you are speaking to the backfit of plants already in operation. Let's try to break it into two phases and see if we can understand it better.

I don't know of a basis for not systematically backfitting the Standard Review Plan to plants in operation other than it would be expensive and a judgment that a good job had been done on safety on those old plants.

Everyone knows that those plants were reviewed by a handful of people compared to the kind of review that is done today. They were reviewed without the operating experience that is available today, and there was no systematic assessment of whether the Standard Review Plan ought to be backfit to them, not in 1975.

There has been since then in the genesis of the systematic evaluation program which does go back and start with the 11 oldest plants to bring them up to some assessment according to current review practices.

Now the plants under construction at the time the Standard Review Plan was issued in 1975 had been reviewed for their CP's in the few years before 1975--1973, 1974, 1975. Since the Standard Review Plan was thought to be a codification of the practice that was going on in those years, then it was judged upon second thought to not be necessary to apply it uniformly and rigorously in the OL reviews for those plants.

I emphasize the word second thought. First thought according to those early office letters in 1975 appears to have been that the Standard Review Plan would be used in the OL review for plants then under construction, but then a subsequent letter said no, we will not do it that way. We will not apply the Standard Review Plan to a whole set of plants, to another set of plants. The burden for justifying the deviations will be on the staff, and then finally down the road a few years, the burden will be on license applicants. Make sure you understand that that burden shifted to license applicants for OL's for the first time in this calendar year, 1979. The first OL's were just filed for whom the burden is on the applicant to justify the deviations.

Q Why place the burden as an initial matter on the staff?

A Well, I think I understand why it was done. I won't defend it, but I think I understand why it was done. First, the Director of NRR said the burden is on the licensees and then the industry reacted and said wait a minute. We don't know the basis for some of your requirements in the Standard Review Plant. They are not written there, and we don't know why you have required certain things. How can we in industry justify a deviation if we don't know what it is you are trying to achieve by the requirement?

Therefore, the burden, the industry said, ought to be on the license reviewers to justify the deviations, at least for those plants that were not reviewed according to the Standard Review Plan at the construction permit stage.

Let me finish. The Director of NRR agreed with that logic and exempted the industry from making that justification of deviations in two stages, placing finally the burden upon the licensees for those plants that are making OL applications here in 1979, that is, those plants that were getting CP's at approximately the same date as first issuance of the Standard Review Plant.

Q Wasn't the NRC at the time that issue came up in the position to explain, to demonstrate the reason why certain things were being included in the SRP on the basis of increased safety?

For example, we have been talking about containment isolation actuation. The TMI 2 scenario has certainly demonstrated I think that containment isolation for 4 PSI in the containment building only is an unsafe way to approach containment isolation, and I take it that is why diverse containment isolation actuation is required under the Standard Review Plant.

A No. Let me interrupt you. It is not unsafe. That is too simple. It is not as reliable. It doesn't cover as many eventualities, as many "what if's," and like many things that we do, judgments have to be reached as to whether even though it is not perfect, it may be good enough.

Now as I testified on the first of June, I doubt that anyone in 1975 specifically concentrated his attention on diverse containment isolation. I have to agree with you.

It looks like something that should have been there. I was surprised it wasn't there in some of the older designs, and it ought to be there. It is a rather minor thing to provide, and it should be there.

I have to believe that if people had concentrated on that specific one in '75 they probably would have backfit it, ~~backfitted~~ but that is not how the decision was reached. It was reached on the whole thing, so the argument on is there a justification or ~~for~~ what is the basis for some of these requirements that the regulators had was a general argument, and it is fair to say generally the Standard Review Plan does not say why the requirements are there. It says what the requirements are. It says how you review to ascertain that they have been met, and it states the findings, and then it gives some reference documents that if you research them, all you would get is a feel for the reason that the requirement was there.

This is not unusual. The Commission's regulations have no statement of basis. That is there is no statement of what they were trying to do, to solve in the regulations, until very recent years similar to the regulatory guides issued by the AEC, and the NRC has no statement of basis until recent years.

Q Do you know if any other plants of the vintage as TMI 2 as they came through the review process were required to have diverse containment isolation actuation?

A I know the answer to the inverse question. There are others that do not have it. I do not know the answer to the question of were there some that had it. My best guess would be since it was codification of existing practice, and I don't believe this specific one was thought of as a ratchet, that there probably are some plants of the same vintage as TMI 2 or even earlier that do have diverse containment isolation.

Q It was a situation where some plants were coming through with that feature, some where not, and the NRC was taking the position pursuant to the provisions you just talked about before that in any event, for the plants of that vintage, that type of diverse actuation would not be required?

A That's right.

Q Okay. If I understood what you said, that was based primarily upon an assessment of the cost of the backfitting versus the increments and safety to be achieved by forcing the utilities to incur that cost and the judgment apparently was made that the provisions of the SRP across the board en masse for some types of plants was simply not worth the trouble and the cost?

A I don't agree with the use of the word cost. I don't believe a cost assessment was done. I think burden is the proper word, assessment of the burden associated with a review and documentation according to the Standard Review Plan as opposed to review and documentation against earlier documentation requirements.

Q What is the one chief objection that the utilities have to assuming that kind of burden?

A I am not quarreling with the fact that the costs were there. I am quarreling with your use of the word costs. Surely costs are the bottom line, but no cost analysis was done by NRC to my knowledge--some general understanding that it was a burden, some requirement certainly existed.

Q To that extent, there was a recognition of cost?

A Yes.

Q I am not talking about a formal cost benefit analysis.

A Okay. That was how I understood you to be using the word. Those cost benefit analyses are done today for backfitting decisions. They weren't done then.

Q I was about to say it actually bothers me more that no cost benefit analysis was done under these circumstances.

If I understand your testimony, it was that the determination was made across the board?

A That is right, to the best of my understanding.

Q All right. Coming back to this question of systems interaction, it is also a subject of some interest as to the question of interaction of operating capability and training, or training vis-a-vis equipment design.

Is there any office within the NRC that looks at the man-machine interface, if you will?

A No.

Q That is not done in the Division of Systems Safety?

A No.

Q Do you view that as a significant lack within the NRC at the present time?

A Definitely.

Q Has that been addressed at all in the lessons learned task force work?

A To a minor extent by our short term recommendations; it is the task force's conviction that this is a major gap in previous regulatory practice and has to be corrected.

We are working on some alternative approaches to address the problem We have not made long-term recommendations in that area yet.

Q You are the team leader on the lessons learned task force, is that right?

A Director.

Q I did want to ask you about that. I have looked over the lessons learned interim report and out of approximately 95 pages in that publication, there is about three and a half pages on operating, training and management, and I was curious as to whether or not that was an area in which the lessons learned task force intends to place a good deal of emphasis or is that considered a secondary matter?

A I think you have given it improper focus. We broke down our work into two principal areas, design and analysis and operations and clearly the operations area after Three Mile Island is an area that is the biggest gap in what we did previously. It may take more words to describe what needs to be done in design and analysis simply because it is of a more detailed engineering nature.

That is the recommendations that we are making at this stage; three and a half pages out of 95 is not fair. That is not a fair assessment of the emphasis that we believe should be placed on the operations aspect of. not only the design, but also the regulatory oversight of nuclear power plants.

Q You do anticipate that in the further work of the lessons learned task force greater emphasis will be placed in that area in terms of further publications or further changes within the NRC?

A Yes.

Q Both within the organization and content of the review and the formulation of requirements, all right.

Dr. Mattson, there has been a great deal of discussion in connection with this investigation by the Presidential Commission of a series of documents that have been referred to as the Michelson reports. There are two handwritten versions of it and one typed version.

Are you familiar with those documents?

A I have read the typed version. I have scanned the handwritten versions. Yes, to that extent, I am familiar with them.

Q Okay. When did you first become aware of the existence of those documents?

A Sometime between the 1st of April, 1979 and the 14th of April, 1979, when I was told in a telephone conversation from Three Mile Island that such documents existed.

Q All right. There has been a report issued by the Office of Inspector and Auditor within the NRC relating to the Michelson report and the events and levels of review concerning that documentation.

Have you read this report?

A I have not seen it. I wasn't aware it had been issued.

Q I have here a copy which I will be happy to make available to you. It is an extra copy. The report purports to quote the contents of interviews conducted with a number of individuals from the NRC, one of them being you, apparently an interview conducted on June 22, 1979. The summary of that interview appears on pages 12 and 13 of the report.

I wanted to ask you about that. If you would turn to page 12 under information obtained from Roger J. Mattson, in the second paragraph, the second to the last sentence in that paragraph, it says that generally describing your beliefs about what impact the Michelson report might have had upon the thinking within the NRC. It stated that, you stress that the intensity of any review would be colored by the manner in which the draft was received, referring to the draft Michelson report in September of 1977. And it states that, keeping with this qualifier, you said the copy of the draft which was informally furnished to Israel, that is Sandy Israel of the NRR, was handwritten, undated, untitled, had no table of contents, and no indicated author, as is indicated in this report. And I was previously under the impression, it seems that the report that was submitted to Sandy Israel in fact did have a title page, a table of contents, a date, and the initials at least of the author, and I was wondering if you had seen some different version.

A Yes. I have the version that was supplied to me by Mr. Israel.

Q I see. Okay.

A I have a copy of it. I think we are probably into an area of document authentication difficulties because there are several versions of this floating around.

Q Right. I am aware of three--the handwritten one in May of '77, the handwritten one in September of '77, and the typewritten one or the typed one in January of '78, but I guess you have got somewhat the third version or the fourth version.

A That is not the right one. I am going to refer to the letter that you have seen because we have discussed it before, which is my letter to Dr. Myers of the Committee on Interior and Insular Affairs of the House of Representatives, the letter of May 24th, 1979, and my recollection as to what I was given in assembling this letter--I may just disprove my own previous words. Let me sort it out here.

Enclosure 2 of that letter is what we are calling here the typewritten version of the Michelson report.

Q That is the one dated January, '78.

A Enclosure 3 of that letter is a draft of that report which was provided to the staff in April of 1979, and that has the initials CM, evidently Carl Michelson, up in the upper righthand corner, a date, a title, and a table of contents.

Enclosure 4 is the draft for the Combustion

Engineering Design, also initialed CPM and dated 5/15/77 in the upper righthand corner.

Enclosure 5 is a set of figures that went with that.

Enclosure 6 is the copy of the report, the draft Michelson report for B&W, and Conbustion, and the figures provided to me by Sandy Israel, and he said this is the information supplied to him by Jesse Ebersole, and it is that document which has no cover, no initials, no author, no date, and in fact you will see that the first page starts at page 3 and goes through. That is the B&W report. If we look at where the Combustion Engineering report starts, it has got a little title up at the top. It has got no title page, no author, no date, and it also starts at page 3, so that is what I meant when I talked to the inspector.

Q Okay. There appears to have been some differences between the September, 1977 report which Mr. Michelson had and the version of that that was transmitted to Sandy Israel.

A At least the first two pages seem to be gone, yes.

Q Looking back again at the Inspector and Auditor's report, on page 13 at the very top it is stated that, "Mattson said that it was his firm belief that formal evaluation of the Michelson report would not have prevented

TMI. He feels that of greater relevance to TMI than formal review of the Michelson report is the fact that applicants can develop and implement emergency operations for conformance with accident analysis reports in the absence of either regulatory review or vender review."

Does that accurately summarize a portion of your interview with the Inspector and Auditor? More importantly, does that accurately reflect your thoughts on the matter?

A Well, it is a very succinct summary of my thoughts, but I will not change them. I believe those things.

Q What did you mean by the second sentence there, the fact that the applicants can develop and implement emergency operations for conformance with accident analysis reports in the absence of either regulatory review or vender review is of greater relevance to TMI?

A Would it help your understanding if after the word operations you inserted the word procedures?

Q Emergency operations procedures? I still am not sure I understand what you meant in that sentence.

A What I mean is that prior to TMI there had been no let me call it technical review of utilities' emergency procedures by the licensing organization of NRC. There had been some review by the Inspection and Enforcement people of several types--one, more technical in nature conducted during plant startup, and another less technical in nature, more of an audit nature during the operations of these plants, but as a general matter, no thorough going, technical review in the licensing process of emergency procedures.

Q What was it about the emergency procedures which you feel was of greater relevance to TMI?

A For example, since the accident at Three Mile Island, we have found that there are or there were in none of the emergency procedures for any of the operating B&W reactors descriptions of the intermittent natural circulation cooling that would follow a very small break loss of coolant accident.

That information has been developed in the form of emergency or operations guidelines and transposed into operating procedures at all of the B&W reactors now as a condition for their restart. That information did not exist in their proceedures before, yet the analytical capability to provide information of that sort certainly existed at B&W prior to Three Mile Island, and potentially had there been regulatory oversight of that aspect of operations prior to Three Mile Island, the staff would have assured that it had been provided to the operations ~~criteria~~ organizations in the form of training or procedures.

Q Is that consideration, would that aspect of operating procedures be related to what appears to have come out of TMI in the form of an over-reliance by operators upon pressurizer level indicator to assess the state of the inventory in the core?

A Not that particular one, no.

Q But you are talking about the mode of circulation within the reactor coolant system, primary system during a small break LOCA, aren't you?

A Yes.

Q A small break LOCA can be such a thing as a PORV sticking open?

A Yes.

Q And under those circumstances as we have seen at TMI -2 you can have a situation where the pressurizer level hangs up, increases perhaps even off scale high?

A That is true.

Q In the context of an operator belief that the pressurizer level indication is an accurate indication of the state of inventory in the core, that would then lead to a misunderstanding as to the current status of circulation within the core?

A I agree. They are indirectly related. It is possible that you could start from a review of the analysis of small break loss of coolant accident in the top of the pressurizer and come to that conclusion.

Q Okay. As a matter of fact, Chairman Hendrie testified before the Senate committee that has been investigating this matter on April 10th, 1979, and he was asked to, I believe he gave a prepared statement as to the six main factors that caused and increased the severity of the accident.

There were two factors that I was interested in. One was the pressurizer electromatic relief valve which opened during the initial pressure surge failed to close when the pressure decreased below the actuation level. This failure was not recognized and the relief line closed for sometime.

That certainly was one crucial factor in the TMI 2, wasn't it?

A That is the small break, yes.

Q Then another one was following rapid depressurization of pressurizer, the pressurizer level indication may have led to erroneous inferences of high level in the reactor coolant system. The pressurizer level indication apparently led the operator to prematurely terminate high pressure injection flow even though substantial voids existed in the reactor coolant system.

That also was a major factor in the TMI 2 scenario, wouldn't you say?

A That is true.

Q Okay. This relates again to the erroneous inferences on the basis of high level in pressurizer level indication?

A Yes.

Q Coming back then to the inspector auditor's report, you do make the statement here that it was your firm belief that formal evaluation of the Michelson report would not have prevented TMI.

Why don't you think that that would have prevented TMI?

A That is why I said it was a little succinct. I did give him more words. We discussed this at some length.

The reason is that formal review of the Michelson report since TMI has led to the conclusion that, A, Michelson was right, and B, the detailed computer codes, the one that B&W uses called CRAFT, were accounting for the features or the phenomena described by Michelson in his report, and that the analysis by those codes of the performance of the emergency core cooling system have accounted for those effects and shown that the ECCS was acceptable. Those analyses would probably have led to a dismissing of the report by Michelson.

If the same mind set that has evidently been here down through the years of not coupling those detailed theoretical analysis with the very practical operating procedures and training had continued to exist when that document was reviewed, I think it may very well have continued to exist.

It took the accident at Three Mile Island to break that mind set. I don't think the single report from Michelson had it been formally reviewed, had it been submitted formally by the ACRS, been charged to people, had it been scheduled and resources assigned to it, all those things that happen for a formal review, I think it would have been the same kind of answer we see with a lot of ECCS-related questions, that is, yes, the codes account for that. Yes, it is okay, the ECCS accommodates that, and the concern is then dismissed because the focus of the report, although it is in there and in hindsight you can see it, the focus of the report was not on what the operator knows and is it proper. I don't believe so.

A These calculations you are talking about by B&W, are those the Appendix K bounding calculations?

A Yes.

Q Isn't one of the basic tenants of the Michelson report that bounding calculations for small break LOCA's are not sufficient to adequately assess the consequences of very small break LOCA's such as the ones addressed in the Michelson report?

A That is true. Maybe my previous answer was too quick to your question. The analyses that have been done subsequently have taken out the traditional conservatism and tried to use the code in a realistic manner, to address the realistic nature of the concerns raised by Michelson, and it is those analyses that show that he is right, but the ECCS has plenty of margin to account for those realistic phenomena. That probably would have happened prior ~~[illegible]~~ to Three Mile Island.

We make realistic analyses and conservative analyses with these codes pretty much as a matter of routine. They are run against experiments and what have you in realistic versions, and then they are run with another deck of cards in conservative versions for licensing calculations.

Q So is your point that even if the Michelson report had received formal review within the NRC and even if the recognition had been made, that pursuant to Mr. Michelson's concern, the bounding calculations for small break LOCA's were not applicable to adequately assess the consequences of very small break LOCA's and even if the further calculations had been done then as suggested by Mr. Michelson, that the conclusion would still have been that the ECCS had the capability to deal with the situation?

A Absolutely. Whether they were conservative or realistic or anything inbetween, the conclusion would have been then as it is today, the ECCS design is fully capable of accomodating that break.

Q I think I understand that point.

A The accident taught us that the procedures for telling the operator how to deal with that break were inadequate.

Q Okay. Let's come to that. Is it your testimony also then that a formal evaluation of the Michelson report would not have indicated to anyone in the NRC with enough responsibility to deal with the situation that there was a strong possibility of operator error based upon a misleading pressurizer level indication?

A I think that is a close call and I wouldn't argue with the judgment either way.

Q Okay.

A It has been my judgment that recognizing the mind set before Three Mile, probably it wouldn't have been realized. However, there is strong evidence that it might have been, and that is the Israel memorandum of January, '78 where out of very general reasoning Mr. Israel came to the same conclusion.

Now had it been subjected to formal review, that is, the Michelson report simultaneously with Israel's general conclusions that led to the Novak memorandum, then it is possible that the judgment would have been reached that more needed to be done in the procedures and training area. I won't quarrel with whichever way the judgment is made. My own judgment is I doubt it.

Q I was beginning with the thought that it was your firm belief that formal evaluation of the Michelson report would not have prevented TMI. However, I can understand why that is a subjective matter.

However, let me direct your attention to the 28th page, and these pages are not numbered. It would be better if you counted from the back, 7 pages from the end of the typed Michelson report of January, 1978, it has a paragraph that begins toward the top of the page with this: "A full pressurizer may convince the operator to trip the HPI pump and watch for a subsequent loss of level." I think you may need to go back another page. I'm sorry. It is several more pages back. I must have fewer pages in mine than you do. It is right above Section 4.6.

Right. You have that sentence there. Then if you look down under Section 4.6 under pressurizer level indication, it says, "The modes of decay heat removal discussed in Section 3.0 point out that pressurizer level is not a correct indicator of water level over the reactor core," and then it follows down a few lines, "Further, therefore, pressurizer level is not considered a reliable guide as to core cooling conditions. No other primary side level indication is provided," and then if you flip over to the next page, skipping down one sentence, it says, "A full pressurizer may convince the operator to trip the HPI pump and watch for a subsequent loss of level; although this response appears desirable, a full pressurizer may not always be a good indicator of high water level in the reactor coolant system."

Mr. Mattson, is there any possibility in your mind that someone with a technical background in the NRC reading that language would not have concluded that Mr. Michelson was expressing a concern about possible operator error based upon pressurizer level?

A I don't quarrel with the statement you have just made at all. What I quarrel with was how would the staff have reviewed and concluded if this thing had been submitted and reviewed formally, and it may very well have been that the staff would have said that the ECCS if allowed to do what it is designed to do, even accounting for single failure, would make this problem go away because there is no opera . role.

Q But there clearly is an operator role.

A There clearly is. There clearly was. It was clearly a role that was carried out incorrectly in the course of the accident, and training would have helped that. No question about that.

The question you are asking me to judge is would we have caught it had we formally reviewed it, and the answer I said, I don't really care whether your answer is yes or no. I know a little bit more about what our mind set was than you do, and I don't want to quarrel with you or anybody who wants to reach the judgment that we would have caught it.

My considered opinion is we may not have caught it.

Q And it was because of a mind set, as you say?

A Yes.

Q I have spent quite a bit of time over the last two weeks deposing people in the NRC, and I understand the pre-TMI and post-TMI thinking.

On the other hand, I am impressed with the fact that you have got some very sophisticated, intelligent people in the Division of Systems Safety, and I read that language, and it only seems to say one thing--possible operator error based on pressurizer level, and I take it you are not really disputing that. It is just you are disputing what significance those words would have carried for someone in the Division of Systems Safety reading that language in 1978?

A Yes.

Q Okay.

A This is A division that historically has not reviewed operator actions and operator training and ~~operator~~ operating procedures, only assured that systems designs be automatic to accomplish their safety functions early after an accident; others in the agency may have caught it, but I don't think the Division of Systems Safety with its pre-TMI 2 orientation,mind set, would have.

Q In terms of operator action, what was that mind set,that the operator would always do what the design called for to bring the reactor under proper control? Is that a fair way of characterizing that mind set as to operator action?

A No. I think a mind set that the operator was a force for good, that if you discounted him, it was a measure of conservatism. If you tried to account for all of the things that he could do, you might have to give away some things in the design because he was a synthesizer and an integrator that went beyond the design, and the oversight that ~~raises~~ implies, of course, is the capability of the operator to intervene in the operation of safety systems.

Now that is why this is a strong argument that this would have been caught in a formal review and a staff would have said wait a minute, that is a difficulty with an operator we never understood before. He will be told to intervene with that safety system. We ought to do something about that.

It may have been that these very clear words in the final version of the Michelson report would have caused that threshold, that trigger level to have been reached in the ~~mind set~~ minds of DSS reviewers.

Q It is possible?

A Possible.

Q Let me take you through--of course, I suppose one point to be made and we should make it, it is my understanding that the report I have just read from the January, 1978 typed Michelson report did not as far as we know find its way into the hands of anyone within the NRC prior to March 28, 1979.

Is that your understanding, the typed Michelson report?

A I don't know about the date you have given me. I don't remember that specific date. I can tell you what my knowledge is.

Q What I had understood was that--

A I first saw a copy of that on April 17th, 1979, by memo from Eisenhut who says that report was given to us, whoever that is, for review and comment earlier this month.

Q Right. You certainly didn't see the typed Michelson report prior to March 28, 1979?

A I never saw it prior to April 17th, 1979.

Q Do you have any knowledge that anyone else within the NRC saw the typed Michelson report prior to March 28, 1979?

A No.

Q You do know, and it seems pretty clearly established now, that the handwritten one, handwritten version of the Michelson report, was passed by Jesse Ebersole to Sandy Israel within the Division of Systems Safety, is that right?

A That is what Mr. Israel has told me.

Q Okay. Let's come to that version which is the version that is dated in September, 1977.

A I don't believe that I have that knowledge. The one that I believe was handed to him is undated.

Q Okay. It appears to be the same from what I can see there except for the absence of the first two pages. It has also some handwriting on top that says the title B&W, but the first paragraph is entitled, "Introduction" and it begins with page 3.

Let's see if it does have some of the same language. Let's flip to page 9 and I am looking at the first full paragraph on that page which says, "Operation in Mode 4,"which is one of the various codes that Mr. Michelson talks about in that report, "...appears reasonable to achieve, although the reactor operator will be unaware of what is happening to the reactor vessel level. Note the presence of a pressurizer level is not an indication that adequate core coverage is being achieved."

Let's flip to page 16 where this paragraph appears. "Adding to these concerns is the uncertainty associated with unknown vessel level, the adequacy of emergency operating instructions, and operator training for this event, and the consequences of the unstable slug flow conditions which are predicted to develop in the piping and safety valves as a consequence of certain operating situations."

These very small break LOCA's considerations appear to be generic for pressurized water reactors, although it may be more severe for B&W 205 fuel assembly plants because of the once through steam generator configuration.

Lastly, let me direct your attention to page 34 where this language appears. "Pressurizer level continues to increase as remaining steam bubble pressurizer cools and is compressed to reactor vessel steam bubble pressure while the hot water drawn from the hot leg pipe into the pressurizer loop seal piping is allowed to cool."

On page 35 in paragraph [illegible] the statement appears, "Reactor vessel level decreases due to fluid loss through the break in excess of makeup capability until makeup capability exceeds fluid loss through break."

Locking at all of that language again, do you feel there is any question that a technical person with a good deal of sophisticated background in the Division of Systems Safety in reading that would not realize what Mr. Michelson was addressing was the possibility of operator error based upon a misleading pressurizer level reading?

A Off the record.

(A discussion was held off the record, and the pending question was read by the reporter.)

THE WITNESS: Yes, I believe that the reviewers in DSS would understand that.

BY MR. KANE:

Q - All right. If you had read that language in 1978, would you have understood that?

A Yes.

Q And presumably if someone in DSS had read that and understood it, would they have taken some action on it to see to it that operators were advised with the correct information as to what to do under these circumstances?

A Yes, but recognizing they would have had to change some things that were then existing review practices.

Q They would recognize that?

A They would recognize that.

Q Would it be customary under those circumstances for whoever recognized that to bring it to the attention of superiors within the NRC so that appropriate instructions could be disseminated to licensees and incorporated into operator training?

A It would probably take a little bit different track. It would take the track of the staff going to licensees to ascertain whether such instructions existed.

Q And if they determine that they did not?

A Requiring that they ~~be~~ generated.

Q I think your testimony before was to the effect it is now being ascertained that those procedures did not cover this situation?

A That's right.

Q I take it then that it is no longer then your firm belief that formal evaluation of the Michelson report would not have prevented TMI?

A May not.

Q If the proper instruction for operators under these type of small break LOCA considerations had been disseminated and if those operators had followed those instructions, would the TMI 2 accident have happened?

A No, not in my judgment.

Q All right. Something I asked you about in our prior interview, Mr. Mattson, related to the Davis-Besse transient of September 24th, 1977. That, as you know, is something that has come up quite a bit, and it has been identified in prior depositions and in some hearings that this Commission has already held as being a precursor of TMI 2.

Are you familiar with the facts relating to that transient?

A I recall the review of the transient. I recall reading some documentation associated with the transient. I didn't personally review it.

Q What do you understand happened in that transient at Davis-Besse on September 24th, '77?

A There was a loss of feedwater. There was a degraded performance of auxiliary feedwater. There was degraded performance of the power operated relief valve. There was voiding in the primary system. There was a fairly rapid recognition of the PORV degraded performance, and a closing of the ~~blocked~~ block valve.

The transient was initiated at 9 percent power. There was a beginning of ~~live~~ life core, not a three-month old core as in the case of Three Mile Island. Consequences to the reactor were none.

Q Was there a PORV that stuck open?

A No. It fluttered. It opened and closed and opened and closed.

Q What was the final result after all that fluttering?

It fluttered and then what did it do? Did it stick open?

A I don't understand the thrust of your question.

Q It is my understanding it fluttered, it opened nine times, fluttered if you will, and then stuck open open.

A That is my understanding also.

Q Okay. It did eventually stick?

A I don't know when it stuck compared to when it was isolated and whether that was significant compared to the coolant loss during the fluttering. I don't know the answer.

Q What did the pressurizer level do?

A I believe it stayed up.

Q Yes. It was on an increasing mode. It dropped slightly, and went into an increasing mode. What did the operator do based on that pressurizer level?

A My understanding is he intervened with the high pressure injection system.

Q Terminated the high pressure injection system three and a half minutes into the event. When did he turn it back on again?

A I don't recall.

Q I don't recall that either. When did he discover the PORV was open and activate the block valve?

A It was a few minutes into the transient. I don't believe it was as soon as three minutes.

Q About 20 minutes into the transient; when did you first become aware of the facts related to this transient?

A In this detail that I have just described, after Three Mile Island.

Q When did you first hear about the transient at all?

A In the fall of 1977.

Q How did that come about?

A It must have been, we are talking about a transient that occurred in September of 1977, late in the month I believe, the 27th of September.

Q The 27th of September you became aware of that transient?

A No. I believe it occurred on the 27th.

Q The 24th.

A So it would be my estimate that sometime in October is when I first became involved, and I became involved when I was asked to attend a meeting held by my staff and the Assistant Director for Reactor Safety.

Q Who was that?

A That was Dr. Ross, who was the AD at the time. Mr. Tedesco ~~was~~ is now the assistant director. ~~of that AD,~~ The meeting was with Mr. Novak and Mr. Mazetis, who was a section leader in the Reactor Systems Group--a meeting with the Office of

Inspection and Enforcement, and the principal representative of that office was Carl Seyfert. The subject of the meeting was to resolve a dispute over who should have the lead responsibility for responding to the Davis-Besse transient.

My staff felt that they should, and the Office of Inspection and Enforcement felt that they should.

Q Before you go any further on that, I am curious because that whole situation strikes me as a gargantuan response to a transient. It would be unusual to have this much involvement over a single transient, wouldn't it?

A No.

Q Really?

A Not a transient of this nature; this is the sort of thing that happens with transients.

Q Was this considered a major transient, a significant transient?

A Significant, yes; I don't know if it was considered major--significant.

Q What was significant about this transient? You pointed out that there was no damage to the plant. It was brought under control in a relatively short period of time. Operator action appears to have been timely in terms of detecting the PORV.

What was significant about the transient?

A Well, I can tell you what was significant in the minds of the people at the time that the NRR staff wanted to be involved because I have a piece of paper that says what was important in their minds, if you will give me a moment to find it.

Q Sure.

A This is a handwritten document entitled "Trip Report." It has got the name Mazetis in the upper right-hand corner. Have you seen it?

Q Yes, as a matter of fact, I have. Why don't you explain on the record what that is?

A Well, it is Jerry Mazetis' summary of his trip to Davis-Besse and his review of the information that was available, and in the summary which begins on page 1 of this trip report, he summarizes what the event was, talks about the system which malfunctioned, and then lists four key areas of concern which I believe must be the things that the technical staff thought were the distinguishing features of this transient which made it an interesting or significant transient as we were just discussing.

Q Where does he list those four key areas?

A The bottom of page 2, there is a list of four, continues on over to page ~~2~~ 3--the auxiliary feedwater system, turbine governors, effect of excessive cooldown rate on primary side. The third is stresses on the steam ~~generation~~ generator, especially steam generator 2 which the applicant believes went dry, and four, dynamic effect of vapor formation in the coolant system during the transient, cavitation and seal effects.

It is interesting that in summarizing he doesn't talk about the level indicator at all.

Q I see.

A And he doesn't talk about steam ~~generator~~ generation, the void formations' possible effect on core cooling. He talks about void formation effects on the reactor coolant pump and its seals.

Q Let me see if I understand. This is a document prepared by Mr. Mazetis at the time he went to Davis-Besse to do an inspection trip in connection with the transient that occurred on September 24, 1977.

Mr. Mazetis has already testified that he took that trip on September 30th, 1977. I believe he has testified that he took it upon direct order or direction from you to go out and inspect that site. Is that true?

A I don't recall. Be careful of the word inspection. I may have asked or it may have been his assistant director. I don't recall. He was asked to go to meet with the people at Davis-Besse and understand the facts of the transient and to interface with the I&E inspectors. Mr. Mazetis is not an inspector.

Q As a matter of fact, at the end of this document, the third and fourth page from the end, are a roster apparently of a meeting that was held on September 30, 1977 at Davis-Besse by Mr. Mazetis with some approximately 32 individuals from Toledo Edison, from NRC, from B&W, and from Bechtel Corporation, and I see that the NRC representatives were Mr. Engel--

A He was the project manager.

Q Mr. Legho is a supervisor engineer in DSS.

A I'm sorry.

Q That is the name that is next to his name.

A I don't know.

Q That is the title next to his name on the list.

A And Mr. Szusiewicz is a reactor engineer, and then there is Jerry Mazetis.

Q Then there is Mr. Ragan, a mechanical engineer from NRC. Then there is Mr. Harpster from Region 3 of NRC, and Mr. Little from Region 3 of NRC.

A That is not an excessive NRC contingent to be present--~~X~~ 2 people from the regional office to make sure they were up to speed on what was happening with the folks from Washington coming out to find out, and Mr. Mazetis who was a reactor systems expert; Mr. ~~[illegible]~~ Szusiewicz who is a containment system branch--I'm sorry, he is Instrument and Control System Branch; Ragan would be the mechanical engineering branch, somebody who understood the thermal stress effects on the steam generator; J. T. Liang ~~Geang~~--I don't know.

Q Okay.

A Engel was the project manager.

Q I wasn't suggesting it was an excessive number. It does strike me, however, that whenever you get 32 people together on a given event, it is recognized that that event has a high degree of significance, particularly under these circumstances.

Wasn't it your understanding at the time that this was a significant transient?

A Yes, but I do not put the word high in front of high significance. We pay attention to a lot of safety things and some of these things take a lot of people. The number 32 doesn't cause me to use the superlative.

Q Is it fairly customary for representatives of DSS to go out on these kinds of trips?

A For significant events, yes.

Q How many times in a year do representatives of DSS go on these kinds of trips, to an on-site situation to sit down in a group of as many as 30 people to evaluate a transient?

A A couple of times a month. I would guess that is roughly what it is, yes.

Q I see that the second factor which was listed out of the four that you said Mr. Mazetis was focusing on was effect of excessive cooldown rate on primary side.

Do you know what he meant by that?

A Yes. The injection of cold water and the boiling off of the secondary side causes the primary side to cool rapidly when it stays ~~are~~ at pressure which creates a potential overstress condition in the primary coolant system in violation of the temperature pressure curves in the Commission's regulations. It is a thermal stress problem.

Q The third aspect was stress on steam generators, especially SG 2, which the applicant believes went dry. I take it he means there that the steam generator boiled dry?

A Yes.

Q Was there any reference to the fact that this was a once through steam generator utilized by B&W in its design?

A That is common knowledge for everyone in the division.

Q Was there focus at this time upon the boilout rate for the once through steam generator in the event of a loss of feedwater as opposed to the boilout rate of the recirculation steam generator?

A No specific focus that I recall.

Q In any event, Mr. Mazetis prepared these notes based on his trip to Davis-Besse on September 30th, and then there was a meeting a few days thereafter?

A Yes.

Q I think you say at that meeting Denny Ross was present?

A I believe Dr. Ross would have been there-- Mr. Novak, Jerry Mazetis, probably Frank Schroeder, my deputy, and me.

Q Was Ashok Thadani also present?

A I wouldn't be surprised. He is with the Reactor Systems Branch, and he is pretty good at these things.

Q How about Thomas Novak?

A Novak is the branch chief. I said he would probably have been there.

Q Was Sandy Israel there?

A May have been.

Q So you had this meeting that was two or three days after this trip by Mr. Mazetis on September 30th?

A I don't recall how many days.

Q Sometime in October of 1977; what was the purpose of that meeting?

A To resolve who had lead responsibility for reviewing the Davis-Besse transient.

Q Was there any discussion at that meeting of Mr. Mazetis' trip report?

A Yes. He presented it.

Q Was it looked over at that time and read?

A In the sense of those are good concerns, those are things we ought to follow up on, those are the kinds of things that the transient raises as questions--the conclusion of the meeting was that I&E would keep the lead and would assure that Mazetis' questions were spoken to.

Q Did you read this trip report at that time?

A I did not recall seeing this trip report until after Three Mile Island. I went to Mr. Mazetis and I said there is going to be some interest in the Davis-Besse transient--I see from Mr. Tedasco's ~~3060,~~ 0560 -- did you keep any document from the time of the review , and I don't recall the conclusion of the concerns that we raised when I&E took the lead and he gave me a copy of this document. That would have been some time in May.

Q You hadn't seen the copy of this document before then?

A Not that I recall; it may have been handed out at the meeting and I didn't retain a copy. I remember the briefing at the meeting spoke to these subjects.

Q If you had been provided a copy of this document, where would you have put it?

A I probably wouldn't. I would have probably thrown it away. I would have listened to the briefing, made the decision that was required of me that day which was to see where the subject was to be assigned for followup, and left it at that.

Q How long did this meeting last?

A Several hours.

Q During the course of that several hour period, it was attempted to analyze the various aspects of the transient to determine who should have lead responsibility for it?

A Yes.

Let me direct your attention to page 6 of this document that we have identified as Mr. Mazetis' trip report. The language appears there "The operator secured ECCS (turned off) HPI pumps and 4 and a half minutes after manual SCRAM he observed a restoration and increasing pressurizer level. At about 20 minutes after the manual SCRAM, the operator concluded that the RV was stuck open and closed a remote manual block valve from the control room,thereby terminating the primary side blowdown."

Do you recall reading that language at the time of this meeting?

A No.

Q Was there any discussion at this meeting of the fact that the operator had secured ECCS, turned off HPI pumps based upon an increasing pressurizer level?

A I don't recall.

Q Was there any discussion about whether or not any operator error did occur in connection with this transient?

A I don't recall.

Q Do you keep a file on Davis-Besse?

A No.

Q During the course of this entire meeting, what was discussed about the transient?

A Well, I recall the conversation focused more on the kind of things that are summarized at page 2 and 3. I remember quite a lot of discussion about the amount of debris generated in the containment sump, and another discussion about the jet impingement on insulation materials caused by the rupture of the disc in the quench tank and the assessment by Mr. Mazetis as to the effect of that quench tank rupture on other systems inside of containment.

As I recall, the meeting was more a hardware oriented meeting, and I recall no discussion of operator error. There may have been some and my memory doesn't serve me.

Q That may have been a function of the same mind set you were talking about before, a mind set which focuses on equipment and design specifications and on design performance rather than upon operator interaction with those designs?

A In this division, yes, sir.

Q You say there was a discussion with Carl Seyfrit. Who is Carl Seyfrit?

A At that time he was a senior manager in the Office of Inspection and Enforcement, headquarters staff. He is now the Director of Region 4 in Dallas I believe.

Q I see. Okay. Mr. Seyfrit was there because there was some question about whether or not this matter should be handled by DSS or by I&E?

A Yes.

Q I am curious in that regard. I have been trying over the last three weeks to get a feel for what the various components in NRC do.

As was reflected in the question I asked you before, my impression was that the central repository, if there is one, for technical expertise in the NRC is largely DSS, and that the capabilities for inspection, for police work, surveillance on what is happening in the field, is with I&E, but that I&E does not have very extensive technical capability.

Is that right?

A I think you underrate their technical capability. They are graduate engineers. They are knowledgeable in nuclear technology. It is fair, however, to say that when they believe that technical problems require a depth of attention or study that is beyond their staff capability, that they transfer the matter to NRR for action.

Now in an operating reactor, that transfer would customarily be to the Division of Operating Reactors. In the case of Davis-Besse, the transfer of the reactor from the Division of Project Management to the Division of Operating Reactors had not occurred, and so DSS was in a bit of an unusual situation. We don't normally review operating experience and assess its significance for the safety of that plant, but since it hadn't transferred, we still had technical support responsibilities in its licensing, and the interface between us and I&E was a somewhat unique interface in this regard.

I was also somewhat unique in having just taken my job, and Mr. Seyfrit and I thought we probably better have a meeting and iron this difficulty out. He didn't think it needed to be transferred to NRR for resolution. My staff had some concerns. They thought it needed to be addressed.

We sat down and decided that my staff's concerns had been well articulated. I&E could factor them into their review and they could handle it.

Q Why didn't Mr. Seyfrit feel that there should be any transfer of responsibility for this away from I&E?

A That his people were capable of addressing the problem.

Q Does I&E have the technical capability to deal with highly unusual types of transients, very much out of the ordinary?

A No.

Q In October of 1977 was the phenomenon of a pressurizer level rising and pressure in the reactor coolant system itself dropping under the circumstances of a very small break LOCA a highly unusual transient?

A Today, yes; then, I am not sure we understood that.

Q But whether or not you understood it at the time, it certainly was highly unusual, wasn't it, in 1977?

A I don't understand how the two statements make sense.

Q I have heard all along that the kind of situation that occurred at TMI 2 was incredible, was beyond anyone's expectation, was something the operators had never seen before or been trained for. It was an extremely unusual situation.

Davis-Besse on September 24, 1977 has in some respects been identified as a precursor of that event. To the extent that it had those features in common with TMI 2 on March 28, 1979, wasn't it a highly unusual transient to that extent?

A Your premise is your statement, not mine, and I am not sure I agree with you there.

Q My question is, is the situation of pressurizer level rising, pressure in the reactor coolant system dropping under the circumstances of a very small break LOCA a highly unusual transient?

A If you understood that those two things were going on simultaneously and that the operator was not trained to understand that they could go on simultaneously, then you would say it is a very unusual transient, but I am not certain, certainly I didn't and I don't think the technical staff at that time attached that significance to it, so it was a significant transient, but more from a standpoint of boiling a steam generator dry and having a rapid cooldown than from deception by the pressurizer level indicator.

Q I am looking again at page 6 of Mr. Mazetis' trip report where he says, "The operator secured ECCS, turned off HPI pumps."

A You have read that once before. I see that, but I also see on page 2 and 3 his summary of the significant features of the transient, and that is not one of them, and that tends to reinforce what I am saying.

The staff recognized that the pressurizer level indicator did what it did while the reactor vessel did what it did, but didn't attach the significance to it that you are attaching to it today.

That may have been an error and poor technical judgment, but it is fact.

Q Let me ask you something a little more specific. Was it recognized at that meeting that the operator had secured ECCS during this transient?

A I suspect it was. I don't specifically recall it; since it is in his summary and the summary was written contemporaneous with the meeting, I assume it was discussed. I don't recall it.

Q All right. Was it also recognized--let me ask you this. Would your response be the same as the subject matter of an increasing pressurizer level during this transient?

A Yes.

Q Okay. To that extent then, with 20-20 hindsight we can recognize it as a highly unusual transient, but I think your testimony is to the effect that was not recognized at the time.

A ~~Usual~~ Unusual connotes low probability. Very significant transient, with 20-20 hindsight, I would agree.

Q To that extent of being highly infrequent as transients go?

A No. Unfortunately, it appears that it is not highly infrequent.

Q It is your understanding that this type of phenomenon of a very small break LOCA pressurizer level increasing, pressure in the primary system dropping, is a fairly usual occurrence?

A Well, one out of every 50 operations of the PORV is the operating history with B&W reactors, as we understand it today.

Q And that has been happening once out of 50 times, that whole phenomenon?

A Yes.

Q In any event, at that time in October, 1977, you say it wasn't recognized as unusual and it was turned over to Carl Seyfrit?

Was it felt at that time that I&E had the capability to adequately assess the technical aspects of this transient?

A With the understanding that Mr. Mazetis was available to advise and consult on the concerns that he had enumerated in the meeting on that day, yes.

Q And you agreed with tha particular decision?

A Yes. I made it.

Q Was any determination made at that time that you should follow up with Mr. Seyfrit to find out what the resolution of these concerns were?

A There must have been. I do not recall the details of it.

Q Did Mr. Seyfrit after this meeting ever report back to you as to his resolution of the concerns involved in the Davis-Besse transient of September 24, 1977?

A I don't recall that he did.

Q Did he report back to anyone else within the Division of Systems Safety?

A I do not know.

Q If I understand correctly, you did not go back to Mr. Seyfrit and say how about it or something to that effect?

A I did not.

Q Would it be customary for you to turn over or has it been customary for you in the Division of Systems Safety to turn over to I&E a technical problem dealing with a transient and to not require some followup to be reported back to you by I&E?

A You missed the point entirely. I&E by definition has an initial responsibility for all operating experience. I&E must make a decision to transfer it back to NRR.

Under ordinary circumstances that would be to the Division of Operating Reactors, never to the Division of Systems Safety directly. We often get problems referred to us by I&E for plants under construction, seldom for plants in operation.

Q On the other hand, you did say that Davis-Besse at that time was still within the Division of Project Management and to that extent then looked to DSS rather than DOR for its technical backup?

A That's right. That is why it was unusual.

Q As to the extent that Mr. Seyfrit would have been doing anything relating to that project, he would have been reporting it back to at least the project manager, wouldn't he?

A He doesn't report to anyone in NRR. It would be his own line organization. Other than preliminary notifications and casual conversation, there is no requirement for him to report any of that back to NRR.

Q Is what you are saying, Mr. Mattson, that after this was turned over to Carl Seyfrit at the end of that meeting as far as you were concerned you were donw with it? You had washed your hands of it?

A That is true, except that it wasn't turned over. It was never taken away from him. It was his responsibility at the beginning, and continues to be his responsibility. Mr. Seyfrit does not report to me.

Q So if I understand it correctly then, a problem which your staff has been called upon to evaluate, with the determination made that I&E has the lead responsibility, at that point, your staff work is done, you have nothing further to do with the problem?

A That's right.

Q And you make no effort to follow up to see how the problem has been resolved?

A That's right.

Q Okay. Do you know today how this problem--

A However, on this particular example, I do recall that Mr. Mazetis was concerned that his concerns be factored into the I&E work. That was the purpose of the meeting. That was the purpose of his summarizing his concerns, and there must have been at the end of that meeting some kind of an agreement as to how Mazetis would provida consulting and other services to the I&E investigators in the conduct of their review. I do not recall what they were.

I find it hard to believe that we walked out of that meeting without agreeing on some feedback mechanism from I&E to the DSS technical staff because they were interested in this transient.

Q Well, it is my understanding from the deposition of Mr. Mazetis which I didn't take that he did have no further contact with Mr. Seyfrit about this particular transient, so I don't know what happened in there, but apparently it went into I&E and did not in any way get back into DSS.

Did it ever go to the ACRS, that we don't know, but we will be taking Mr. Seyfrit's deposition tomorrow.

A Someone told me that there was a presentation by I&E to the ACRS on the Davis-Besse transient which means if there was, it is in a public record transcribed somewhere.

Q That was a presentation to the ACRS concerning the September 24, '77 transient?

A That is my recollection as to what I was told within the last several months by someone that I don't recall.

Q Do you know when that presentation was made to the ACRS?

A I would think it would have been in either the October meeting or the November meeting is where I would look.

Q Of 1977?

A Yes. If I didn't find it there, I guess I would ask the ACRS staff.

Q There is this meeting then about the Davis-Besse transient in early October of 1977, at which time they confirm that Mr. Seyfrit will have lead responsibility.

Mr. Ebersole has testified that it was also just about this time in October, '77 that he gave to Sandy Israel the handwritten copy of the Michelson report we have been discussing, and it is my understanding that this Michelson report in part engendered the Novak memorandum that you made reference to before dated January 10, 1978.

A The Michelson report did?

Q Yes. I believe Sandy Israel's testimony at this point is to the effect that he is not sure whether or not the--

A I was going to say that Sandy told me several months ago that he tended to think it was the Davis-Besse transient that led to the memo, but it may have been the Michelson report. They occurred approximately the same time.

Q As a matter of fact, he makes reference to the Davis-Besse memorandum in the Novak memorandum or the Davis-Besse transient. It is my understanding that Sandy Israel drafted the Novak memorandum for Novak's signature.

Is that your understanding?

A That is my understanding.

Q This is a document dated January 10, 1978. It has been previously attached as Exhibit No. 3 to the deposition of Sandy Israel.

Is this the document you have seen which you recognize as the Novak memorandum?

A It is. It has got my handwriting on the bottom. It says Enclosure 7. That would have been the enclosure to my letter to Henry Myers.

Q And if I understood your testimony before, do you feel that this memorandum sort of hits it on the head as to what happened to TMI 2 on March 28, 1979?

A It certainly describes what happened to the pressurizer level indicator.

Q Does it also address the question of operator error based upon that pressurizer level?

A You will have to give it back to me.

Q I would specifically refer your attention to the second to the last paragraph where this language appears. "Although the safety analyses do not require termination of the makeup system, operators would control makeup flow based upon the pressurizer level as part of their normal procedures. As a result, under certain conditions where the pressurizer could behave as a manometer, the operator could erroneously shut off makeup flow when significant void occurs elsewhere in the system or loss of inventory is continuing."

A It certainly does.

Q It appears to address operator error based on pressurizer level, doesn't it?

A Yes. You know, that is an anomaly in this memorandum. It stresses operator error, but then look what it does in the next paragraph, and there is the mind set again. What does it say to do? The basis for the design requirement be studied carefully for all CP reviews for the object of determining if the loop seal can be eliminated. OL review procedures should be reviewed.

Q To ensure adequate information before the operator terminates makeup flow.

A No procedures are reviewed in the Reactor Systems Branch, and so the emphasis is again in the design.

Q Where are procedures reviewed?

A The kind of review that ought to be done, nowhere; they are not reviewed by the industry. They are not reviewed by the utility. They are not reviewed by us. They are being reviewed right now, and it is a start in the right direction.

Q I realize I am approaching this as a layman, relatively unsophisticated in these areas, and certainly unsophisticated by the NRC and where it has come from, but the very idea that NRC licenses mechanical devices which are out there in the country to perform, and does not examine the procedures to be utilized in operating those devices strikes me as very anomalous at the very least.

How did that come about?

A You have said it too all encompassing; first of all, there is some review, not of the type described here, but there is some review.

Q What is that review?

A Review to ascertain that the procedures for the anticipated events, the transients, and the accidents described in Chapter 15 of the safety analysis report, exist.

Q Now how they are implemented, just that they exist?

A Let me finish. That they exist; second, during startup of each reactor, the Office of Inspection and Enforcement goes through those procedures, and I suspect there is ~~are~~ some spectrum in ~~of~~ the goodness of that quality assurance review, but they do go through the procedures; and in the past there has been one additional point of inter~~f~~action between the procedures and NRC, and that has been the use of the procedures by the operator licensing staff of NRC in the conduct of its examinations of opeators.

Q That would be--

A The kind of review that has not been done, is to take the detailed systems reviewers and designers and analyzers of the type that exist in DSS and couple them with the people oriented reviewers that exist in the Operator Licensing Branch and in the Office of Inspection and Enforcement. That is a mistake, that is, but it is a fact.

Q Okay. If I understand it correctly, if we are talking about a fairly esoteric technical question, one that really has not been considered before at PSAR or FSAR or a given project, or that has not been addressed in terms of the on-going work on unresolved generic safety issues, if we are talking about that kind of issue, something really new, it is in DSS that it is likely to get the most sophisticated treatment, is that right?

A Yes.

Q So if you had a new idea, something that really hadn't been focused on before concerning pressurizer level and how it functions under certain types of small break LOCA conditions, it would be to DSS that you would want to address that issue for evaluation, wouldn't you?

A Yes.

Q Okay. In fact, in having the Michelson report pass Sandy Israel, in having Sandy Israel prepare this Novak memorandum that we have been talking about, wasn't this issue of the behavior of pressurizer level indication under certain small break LOCA conditions presented to DSS?

A Yes.

Q What happened to that presentation after it came to DSS?

A I think you have already said what happened.

Q If I understand it, the Novak memorandum did not go beyond the Reactor Systems Branch. What we are able to ascertain is that he simply went into the files of the various respective people who received it. They kind of mentally earmarked it for possible use in further reviews of B&W plants, and that's all.

Is that your understanding?

A With the exception of Mr. Ignatonis who was the reviewer on the Sun Desert construction permit application He phrased a question for that Westinghouse design along the same lines to ascertain (I suspect all that, I haven't spoken to him) whether the same kind of problem could exist in the Westinghouse design.

That question went to the Division of Project Management and was not mailed to Sun Desert, ~~and that~~ the plant was cancelled.

Q That project, as I understand it, never went anywhere because of other problems that came up in California relating to the construction?

A The project was cancelled, yes.

Q Okay. However, in all of that activity within the Reactor Systems Branch relating to the Novak memorandum, nowhere do I find any recognition that the problem addressed in the Novak memorandum or the problem addressed in the Michelson report was of generic concern. Is that right?

A The Novak memorandum on the next to the last sentence recommended that the design requirements be studied carefully for all CP reviews.

Q Doesn't that mean if that is a generic recognition, it also applies for existing nuclear power plants?

A It does to me.

Q It apparently did not to the people that received this at the time, if you can judge their thoughts from their actions since there was no word put out to DOR or to existing licensees concerning this matter at this time, was there?

A Not to my knowledge.

Q In fact, the memorandum, the Michelson report that we made reference to before states also that it appears that these very small break LOCA conditions are generic to pressurized water reactors, although it may be more severe in the case of B&W plants.

What is your understanding today as to why the generic implications in that sense of these concerns raised in the Novak memorandum, in the Michelson report, were not recognized and acted upon by DSS at the time?

A I believe that the technical people who had knowledge of the things you have described judged the effect to be a small one perhaps because of their conviction that for a small break loss of coolant accident there was a reliable emergency core cooling system which required no operator action of the sort that would depend upon pressurizer level.

Q There was simply no recognition of that you feel?

A Yes. I must say that I have read the internal memoranda from B&W which speak to the same point evidently from about the same time period, and they reach a somewhat different conclusion than that, and so in giving my answer, I think I am saying that the people of the NRC staff were not as insightful on this particular problem as the people at B&W were.

Q When you say that you say you have seen some B&W calculations which have indicated they did consider it--

A No, no. I am speaking of the Bert Dunn memoranda.

Q We know how B&W handled that situation.

A I am simply saying that the B&W analysts comparable to the DSS analysts who apparently were studying the same problem at the same time, understood it differently than the DSS analysts.

Q I wondered about that because you do make reference to the B&W analyst, and I am aware that Mr. Dunn of B&W did recognize this problem, but I am curious in that the Michelson report, the handwritten version which was provided to Sandy Israel by Jesse Ebersole in October of 1977 has attached to it two appendices, those being Appendix B, which is small break LOCA analysis by Babcock and Wilcox dated in March of 1976, and then it has attached as an Appendix C the NRC evaluation of B&W small break LOCA analysis.

Now I can tell you that we have deposed Mr. Michelson, and we have gone through this entire report with him and have asked him about these appendices. He assures me that neither Appendix B or Appendix C, that is the B&W analysis, which speaks of these concerns, and the NRC evaluation of B&W's analysis in any way address his concerns.

Now again, is there any way that this could have been provided to DSS, that it could have been redundant and yet not understood that these considerations were not at that time at least being addressec by either B&W or the NRC?

A Well, I don't understand your question at all. The report you refer to by B&W is written in March, 1976, a year and a half before the draft Michelson report was handed to Sandy Israel, and the summary and conclusions page is dated 1975, which is even older, and Appendix C is written in January, 1975.

Q '76.

A You're right, 1976--I don't see how they could have been expected to address Mr. Michelson's concern if he was looking for an answer to the specific things he said.

Q It is my understanding from Mr. Michelson that as of September, '77, this was the best existant evaluation that he could fine that approached the kind of concerns he was addressing in the September, '77 handwritten memorandum.

A I suspect that is true.

Q Okay. My only question is how could it be possible for someone in DSS in reading that document, including the appendices, to not realize that these concerns being addressed by Mr. Michelson had not been addressed by B&W and the NRC in its prior analyses of the problems?

In other words, isn't it an inevitable conclusion in looking over this report that these are new concerns?

A Yes.

Q Okay. Something else that I wanted to ask you about in connection with pressurizer level at this meeting that was held in early October relating to the Davis-Besse transient, was there any discussion of ECCS actuation?

A You asked me that already.

Q Right.

A I said there must have been because it is part of the summary, but I don't recall it.

Q I didn't ask you that. I asked you about operator error in connection with ECCS.

A You asked me about both I think.

Q Was there any discussion about coincident logic for ECCS actuation?

A Not that I recall.

Q Are you aware that coincident logic is utilized at a large number of operating nuclear power plants in the United States for ECCS actuation?

A Not any more.

Q Right, but as of March 28, 1979 I am informed there were approximately 25 such plants, is that right?

A However many Westinghouse designs there were.

Q This is a common Westinghouse feature?

A Yes.

Q If I understand coincident logic, at least in connection with ECCS actuation, one facet of it is that it is often tied to pressurizer level and pressure in the reactor coolant system such that the ECCS will not actuate unless a set point for pressurizer level and a set point for pressure in the reactor primary system are reached, is that right?

A Basically right.

Q So that in the circumstances of the phenomena pressurizer level goes high or stays high and pressure in the reactor coolant system drops, the ECCS will not automatically actuate?

A That could be.

Q And those would be circumstances under which you would very much want the ECCS to automatically actuate?

A That is true.

Q In connection with this meeting that was held in early October dealing with the Davis-Besse transient of September 24, '77, was there any recognition that the phenomena being looked at at that time posed a problem for coincident logic ECCS actuation?

A No. As a matter of fact--no, not that I recall, and as a matter of fact, to go to one step further, I don't believe that there was that problem in many people's minds even in the first few weeks or months after Three Mile Island because the pressurizer level difficulty with the B&W design was being largely attributed to the loop seal arrangement in the B&W design, but in the course of the last month or so, the possibility of countercurrent steam water flow problems in the pressurizer for both Westinghouse and Combustion Engineering designs has been shown to be another method for causing the pressurizer level to hang up for a break in the steam space in the pressurizer, a small break.

Q Are you saying it was thought at first that this problem of the loop seal would be applicable only to B&W plants and would not pose a problem for Westinghouse or CE plants?

A That's right.

Q All right. Wasn't it recognized that totally aside from loop seals, you could have the kind of small break LOCA phenomenon described by Mr. Michelson in his reports occurring at Westinghouse and CE plants as well as B&W plants?

A Yes, but the deception in the pressurizer level indicator was not appreciated from that standpoint.

Q Again, I hate to keep coming back to it, but page 16 of the handwritten report provided by Jesse Ebersole to Sandy Israel makes reference to the fact that these very small break LOCA considerations appear to be generic for pressurized water reactors, although the problem may be more severe for B&W plants.

A You are reading that very narrowly, and I don't believe that statement is intended to be read narrowly. That means this general problem of more decay heat being generated than can be removed by the discharge out the break, not necessarily the very specific thing of an anomalous pressurizer level indicator.

There are a couple gems of wisdom in the Michelson report and you have got to be careful which one you are reading about.

Q That sentence immediately comes after a sentence which says, "Adding to these concerns is the uncertainty associated with unknown vessel level, the adequacy of emergency operating instructions, and operator training for this event, and the consequences of unstable slug flow conditions which are predicted to develop," et cetera.

It is talking about operator training, emergency operator instructions, and the uncertainty associated with unknown vessel level based upon a misleading pressurizer level reading.

Now that is what the report seems to be referring to, so when he talks about the fact it is generic to pressurized water reactors, although it may be more problematic for B&W reactors, isn't he making it clear, Mr. Michelson, that he is concerned about this problem across the board, not just at B&W reactors?

A That is true.

Q Okay, so it would appear to be applicable to the Westinghouse situation?

A That is true.

Q Again, if I understand your answer, it is simply that the implications of this hangup in pressurizer level were simply not appreciated at the time the Davis-Besse transient was being evaluated in terms of who was going to have responsibility?

A I think they must have been very shortly thereafter because that meeting was early October evidently, and the Israel draft of the Novak memorandum of January 10, 1978, two or three months later that he is writing about it; I think there must have been such a connection.

Q Made between the pressurizer level and pressure performance and the problems with coincident logic ECCS actuation?

A No.

Q That was my question.

A Okay. To go back to that specific thing, evidently it was not, that connection was not made until after Three Mile.

Q Does I&E have the technical capability to make that kind of connection?

A Yes, if they see it clearly, but evidently they not only didn't see it clearly, but we didn't see it clearly because we didn't make the connection here.

Q Is I&E in the business of trying to see those kinds of connections? The reason I ask that is because we have spent some time examining people from I&E, and the impression I get is that I&E is very much burdened with very specific and very routine types of specifications, regulations, requirements, that must be met by the licensees, and that when they go out to do an inspection, they are by and large devoted to making sure that those requirements have been met. They have little time, if any, for creative thinking or creative connections among safety problems.

Is that correct?

A I know people in I&E whose minds don't work the way you have described and who are quite creative. As a general matter, I suspect that what you have described is closer to the norm than what I would characterize as what some I&E inspectors have demonstrated to me that they are capable of doing.

Q Again as a corollary to that, that to the extent that a problem would be new, something not experienced before with any degree of frequency, it would be DSS and not I&E that would have the real technical sophistication to fully evaluate that situation, wouldn't it?

A For the operating experience brought to our attention, yes.

Q All right. Fine. In any event, I gather the upshot of this October meeting was that the matter was entrusted or was left I guess is the correct word with Mr. Carl Seyfrit and I&E, and that there were no specific arrangements that you can recall being made at that time for Mr. Seyfrit to report back to DSS or to let you know what he had done, is that right?

A That's right.

Q Do you have a personal opinion as to Mr. Seyfrit's competence to address generic safety issues relating to nuclear reactors?

A No. May we take a break?

MR. KANE: Sure. Let's take a short recess.

(A brief recess was taken.)

BY MR. KANE:

Q Mr. Mattson, I want to come back again to Mr. Seyfrit just briefly since he was the one to whom this matter of the Davis-Besse transient was, with whom it was left.

At the conclusion of your meeting in October, '77 there was a transient on March 29th, 1978, after this meeting on Davis-Besse, after the Michelson report is gone into and Mr. Israel, after Mr. Israel has generated the Novak memorandum, and then this transient in March of 1978, a PORV stuck open, and at that time there was an evaluation of what was necessary to prevent operator ignorance as to the position of the PORV, there was no indicator available at that time on the control board at TMI 2 for the position of the PORV. Do you recall that transient?

A No. I recall the discussions of it since the Three Mil^ Accident, but I recall no knowledge of that transient prior to the accident.

Q Was it recognized in March of 1978 as common knowledge that pressurized water reactors utilize PORV's?

A Oh, yes.

Q That was why the note. Was it widely know there was not a position indicator for the operator in most control rooms relating to the PORV?

A Had I been asked prior to the accident at the Three Mile Island, I would have not known the answer to that question. I don't know what was in the minds of the staff.

Q Isn't the reason you would not have known that that the PORV is not a safety-related piece of equipment and therefore is not something that DSS focuses upon?

A That is probably the reason, yes.

Q I would like to cover that point with you.

A I may also be giving my answer to that question that I personally hadn't ever focused on that element of the design.

Q But DSS only focuses on safety-related aspects of the design, doesn't it?

A That is true.

Q It doesn't get into the other stuff in the plant?

A That's right.

Q So as a matter of professional assignment you wouldn't get into PORV's either?

A No, not necessarily, because I have looked at PORV's from a safety standpoint as I was in that time period, and that would be in connection with their use in the event of an anticipated transient without SCRAM, the unresolved generic issue.

Q In connection with that, did you look at the situation of the reliability of PORV's?

A No.

Q Did you look at the question of PORV's occasionally sticking open?

A Yes.

Q Did you look into the history at that time of how often PORV's had stuck open in the past?

A No, and maybe I answered too quickly. We were looking, in that time period, my staff was looking in that time period at the capability of PORV's to withstand slug flow and whether they could be relied upon for high pressure two phase and solid water discharge in the event of an anticipated transient without SCRAM.

We had not looked at them, to my recollection, from the standpoint of their function in normal service, but rather their utility and reliability for this beyond normal service application. That is,what mitigating capability would they bring to anticipated transients without SCRAM?

Q What timeframe are we talking about for this examination by your office?

A Of ATWS, starting about 1970.

Q About 1970, so during this 1977 peri:d when you had this meeting about Davis-Besse and on through the Michalson report being transferred to Sandy Israel and the generation of the Novak memorandum, and even up to the time of this transient of March 29, 1978 at TMI 2 with the PORV sticking open, your office was looking in on an on-going basis, into anticipated transients without SCRAM and in that connection with the performance of PORV's in some respects?

A Yes.

Q In any of that evaluation was there any consideration of how the operator would tell what the position of the PORV was under any given circumstances?

A Not to my recollection.

Q Something we have talked about before is this distinction which comes up again and again in conversations I have had with persons within the NRC on safety-related equipment and non-safety-related equipment, and Appendix B to 10 CFR, Part 55, the first time that was brought to my attention--

A Not 55.

Q Part 50, excuse me--the first time that was brought to my attention I thought I could go to Appendix B and find a listing of each little valve and each little pin and whatever that is safety related.

Of course, I found out that is not the case. I gather Appendix B is a broader, more general definitional type of guidance for the licensees, and if I understand correctly where the licensing process works is that the licensee in the PSAR designates those items in the plant that are considered safety related. Those items are then reviewed by the NRC and presumably at some point approved.

Is that right?

A That is true.

Q Is the PORV a safety-related device?

A It has been judged not to be a safety-related device.

Q I am curious about that. It was also my understanding it was judged in the past that anything that constitutes the boundary of the primary pressure system is safety related.

A That is true.

Q And as I understand it, in normal configuration, the PORV constitutes a portion of the boundary of the primary system.

A That part of it has to meet the code requirements, the ASME code requirements, for pressure retaining capability, but the functioning of the valve itself -- opening and closing and its operability -- is not a safety function, only its pressure retaining capability, so for example, its body must meet the requirements of the ASME code, but its actuator and position indicator and those sorts of things ~~do~~ need not meet the other requirements such as single failure, diversity, redundancy, testability, seismic capability, so on and so forth.

Q I am curious about that. Why is it that anything that forms a portion of the boundary of the primary pressure system is safety related? What is it about that type of thing that makes it safety related?

A The answer to me is obvious. Let me pause for a minute.

Q I think it is obvious to me, too, but I would like to get your answer.

A Because the primary boundary, that is, the reactor vessel and the primary piping, is a safety function.

It is necessary to continue the cooling of the core.

Q And it is also necessary to prevent radioactive water from leaking out?

A Right, yes.

Q If the PORV jams open, you are having radioactive water leaking out somewhere out of the primary pressure system, aren't you, so under those circumstances, wouldn't the performance of the PORV be a safety-related consideration?

A Yes. That is an interesting way to come at the defintion, and there are people who are coming at it that way. It was not the way that it was come at before. It was not the way we looked at it before.

Before we would have said there are three alyers of defense in the sense of keeping radioactivity away from people, the first being the fuel cladding, the second being the primary boundary, and the third being the containment boundary, and for anticipated moderate frequency events, the requirement was that the first boundary, that is, the cladding, not be violated.

For accidents, the requirement was that the primary boundary not be violated except those accidents that were by definition a violation of the primary boundary. That is a failure of the primary boundary, in which case the requirement was that the containment not be violated up through a certain class of design accidents, ~~not~~ including all design basis accidents, but ~~and~~ not including all possible accidents.

Now your logic says, and I think it is probably a good logic but it wasn't the logic that was applied before, your logic says that you preserve each boundary for any kind of event. That is a non-mechanistic approach which is in theory the better approach to defense in depth as we have had it through the years.

It turned out that in its application we haven't been that non-mechanistic. We have tended to be more mechanistic, so mechanistically we would say there is a block valve on the PORV, and if ~~it~~ the PORV fails ~~to~~ open, then the block valve can be closed.

Unfortunately, we didn't follow the mechanistic argument through to its logical conclusion which would have said then is there proper indication? It is reliable? It is redundant? It is safety grade? Instead the line between safety grade and non-safety grade was interposed there for some historical reasons that I don't know and I doubt are documented anywhere.

Q The line being there was a block valve which could be activated in case it jammed open. Doesn't that mean the block valve has to be safety related and safety grade because now that is the primary boundary that you are relying upon in the event the PORV jams open?

A In a sense of its pressure retaining capability, it is.

Q Safety related?

A Yes.

Q What about in terms of a position indication for the block valve?

A Well, what about its emergency power? The line was drawn in a poor way.

Q If I understand it, what you just told me,the line was not drawn at the PORV and its performance because of it jammed open, you had a block valve.

On the other hand, if the full operation of the block valve was not considered safety related, and I take it that is because there was a PORV?

A Yes.

Q That is right?

A That is my best understanding of what must have been historical reasons why PORV's and block valves seem to have escaped proper categorization and review and what have you as safety equipment.

Q Do you think if it had been safety related, there would have been more attention to the generic implications of problems with PORV's?

A Oh, yes.

Q I want to come back to that March 29, 1978 transient at Davis-Besse.

A March 28th?

Q The 29th, 1978, one year to the day I guess, or less a day.

A Not at Davis-Besse.

Q I'm sorry. I mean Three Mile Island, March 29, 1978; it is my understanding that there was an action item control form submitted by Metropolitan Edison, or excuse me, submitted by I think this must have been done internal at the NRC, concerning a review of the adequacy of the design approach in connection with the PORV at Three Mile Island.

I have here a document which is entitled "Action Item Control Form" and I want to ask if you have ever seen that before?

A No.

Q Do you know what that is?

A No--I have never seen the form before.

Q I see. It does appear to be something internally used within the NRC to deal with these problems.

A If I read it, perhaps I could tell you what it is.

Q Sure. Okay.

(The witness read the document.)

THE WITNESS: It is assigned to a man by the name of Woodruff in the Office of Inspection and Enforcement.

Having not seen it, I must conclude that it is an I&E internal form, and it refers to a memo to Seyfrit, to SN I&E. I don't see any indication here that it comes to the Office of Nuclear Reactor Regulation.

BY MR. KANE:

Q I see. Okay.

A But there may be. There is some coding.

Q It seems to involve Mr. Sternberg as well.

A I don't know that name.

Q He is with I&E. There was a follow up this memorandum which is dated March 31, 1978. This is a memorandum for Mr. Seyfrit through Mr. McCabe from Mr. Sternberg, all within I guess this is Region 1.

Mr. McCabe is Region 1, and Mr. Seyfrit is I&E headquarters. It refers to this event which occurred on March 29, 1978. It states that the cause of the reactor trip was the loss of a vital bus caused by an inverter failure, and the cause of the inverter failure is understress by the licensee.

It did, however, relate to this blowdown which was caused by a PORV opening, and it states that the relief valve does not appear to be safety grade component. It requests that the adequacies of the design approach be reviewed on an expedited basis, the design approach being identified as valve failing open and loss of control power.

Have you seen that document before?

A No.

Q You know what a bus is, of course?

A Yes.

Q Until I started this investigation, I thought a bus was something you rode to work, but it is my understanding now that it is a type of electrical circuitry?

A Yes.

Q If I understand what the situation was at TMI 2 on March 29, 1978, it was that as long as energy was in the bus relating to this PORV, the PORV remained closed. If the energy failed, power was lost to the bus, the PORV would fail open.

Does that sound right?

A Sounds like a very interesting design. I was not aware of that.

Q It strikes me as a bad design to the extent that what it means is that if power is lost, the PORV fails open. Does that strike you that way also?

A Yes. It is not a good design.

Q It is also my understanding that as a result of this event of March 29, 1978, the circuitry was reversed at TMI 2 such that it took power in the circuitry to open the PORV and that when power was lost, it became de-energized. Then the PORV would remain closed.

Are you familiar with that situation at TMI 2?

A No.

Q Okay. That strikes you as a better design, though.

A Yes. Off the top of my head it does. I really shouldn't reach judgments that quickly. There may be situations where that wasn't good,but I am not aware of any.

Q All right. Do you know if that type of circuitry I just described in unchanged state where it requires energy to keep the PORV closed and if energy is lost to the circuitry the PORV fails open, do you know if that kind of circuitry is the circuitry that is followed for PORV's in operating reactors around the country now?

A I do not.

Q If that were the situation,you would have some concerns?

A I certainly would look at it, yes.

Q There was a further memorandum prepared here in connection with this incident of March 29, 1978 at TMI 2 from Mr. Seyfrit to Mr. Brunner.

Do you know who Mr. Brunner is?

A No.

Q He is the Chief of the Reactor Operations in the Nuclear Support Branch in Region 1. It references that by memorandum dated March 31, 1978, which is the second memorandum I showed you, the region requested that we review the adequacy of the control system for the subject valve and it refers to the TMI 2 pressurizer relief valve.

It states that failure in this position, that is, failing open position, is covered in Section 7.4.1.1.6 of the FSAR. "We conclude that additional review is not warranted."

Have you seen that document?

A No.

Q Okay. That is apparently Mr. Seyfrit's determination that no further review of the problem is required. It is my understanding that at that time, the command signal indicator was installed at TMI 2 on the control board, and that that was considered then to be an appropriate solution for the problem along with the reversal of the circuitry.

What I want to ask is why to your knowledge was there no appreciation of the generic considerations of this problem relating to PCRV indication?

A I have no idea.

Q It does appear that Mr. Seyfrit did not appreciate the generic considerations of the problem, does it not?

A It appears that he took no action on the generic implications in these memoranda.

Q Okay. Based on what you know today, are there generic implications concerning PORV indicators for the position of the PORV?

A Definitely. There are also safety implications about the failure mode.

Q Mr. Mattson, are you aware of any transients which have occurred outside of the United States involving nuclear reactor plants in which PORV's have stuck open?

A One.

Q When was that one that you are aware of?

MR. CHOPKO: Objection.

THE WITNESS: You are worried that this is classified?

MR. KANE: There has been come concern in that regard. We are anxious, however, to get the information if we can. We have been informed that the transient was in 1974.

MR. CHOPKO: Objection.

THE WITNESS: Somebody told me today that--I can satisfy your concern. There has been a public version of a report describing the events to which you refer just recently made available to your people, or about to be made available to your people.

MR. KANE: We don't have any problem on this then?

THE WITNESS: I am not sure whether it answers your question.

MR. REILLY: I think it is just that one statement. I don't know.

MR. KANE: All I am asking is when did the transient occur.

MR. CHOPKO: Until we see the statement, I would prefer not to have anything on the record at all to avoid an unauthorized disclosure of classified information.

THE WITNESS: I am perfectly willing to discuss that with you, but I am bound by restrictions on classified information.

All I know about that transient is contained in classified memoranda, classified by others; by law I am required, Kevin, to honor that.

MR. KANE: Okay.

BY MR. KANE:

Q When did you find about about this transient?

A Since Three Mile Island.

Q How did you find out?

A Mr. Thadani came to me in late spring, early summer, in the context of his work on the bulletins and orders task force. He normally works in the Reactor Systems Branch and reports to me.

Q He told you about this transient?

A He told me they had just learned of such an event. He sought my counsel on how to proceed with gaining further information. I gave it to him.

Q Was this transient at a Westinghouse plant?

A I don't know what is classified any more on this.

MR. CHOPKO: That particular one is not classified.

THE WITNESS: If he advises me that it is not classified, then I will go with that. It was a Westinghouse.

BY MR. KANE:

Q When did Westinghouse find out about this transient?

A I don't know.

Q Do you have any information as to whether or not Westinghouse was aware of this transient prior to March 28, 1979?

A I have a lingering suspicion, but I have no direct information.

Q Why don't you give me your lingering suspicion.

A Yes.

Q What leads you to that lingering suspicion?

A I recall being told that there were reports not available to the AEC or NRC, but describing this event, that were in the United States.

Q I'm sorry. Could you repeat that?

A There was a report describing this event that was in the United States.

Q Who submitted that report?

A I don't know.

Q Who told you about this report?

A It must have been Mr. Thadani and the people in the task force, the bulletins and orders task force who were researching this event.

Q This is a report dated prior to March 28, 1978?

A It is my recollection that there was written information of some sort available in the United States prior to March 28, 1979.

Q Was that information in the hands of the NRC?

A I said no.

Q I'm sorry. I didn't hear that. Who had that written information?

A I don't know that either.

Q Who had that written information prior to March 28, 1979?

A I do not know that. I said I had a suspicion.

Q Okay. What form did that written information take? Was it a report of some kind or a memorandum or what?

A My recollection may fail me entirely here, but I recall someone mentioning to me about the time I first learned of this event, late this spring or early this summer, that a report existed.

Q Do you know who prepared that report?

A No.

Q Do you have any reason to think it was a Westinghouse report?

A No.

Q Do you have any idea how or when this report was available?

A No.

Q Except that you know it was available or you have heard it was available prior to March 28, 1979?

A Yes.

Q Your best recollection as to who told you is that it was Thadani?

A That's where I would start.

(A discussion was held off the record.)

MR. KANE: Back on the record.

BY MR. KANE:

Q It is my understanding from previous testimony we have taken, Mr. Mattson, and also some references you have made, that coincident logic ECCS actuation has been in the past a common feature of Westinghouse plants, is that right?

A That is true.

Q Would knowledge of the full details of the transient we have just been discussing prior to March 28, 1979 have been of material assistance to the NRC in evaluating the safety of coincident logic ECCS actuation?

MR. CHOPKO: I am going to object to that. It relies on assumptions some of which are going to be classified.

MR. KANE: I am not asking for him to disclose the assumptions.

MR. CHOPKO: Your question assumes it. I will object to the form.

MR. KANE: All my question is directed to is would the full knowledge of those facts which have not been disclosed on the record here have assisted the NRC in evaluating the safety of coincident logic ECCS actuation as a general design feature?

MR. CHOPKO: I am going to object.

MR. KANE: I am not relating it to any specific plant, just as a general design feature.

MR. CHOPKO: Renewed objection.

MR. KANE: I don't know what the objection is based upon. I am not asking for any specific information about anything that is covered by any confidentiality agreement.

MR. CHOPKO: You are asking based on knowledge of this overseas event, and you are linking it to specfic factual information, and you are basing that again on assumptions.

MR. KANE: I am not linking it to any specific factual information because the specific factual information has not been disclosed here.

THE WITNESS: I appreciate what you are saying, and I appreciate your side of the argument. Let me give my answer because I think it moots the point.

I don't know. I would have to review the information and look at again coincident logic. It is not a question I asked myself when I saw the information, and I don't know the answer to your question.

BY MR. KANE:

Q Is it your understanding that the situation relating to coincident logic ECCS actuation has now been changed at operating reactors in the United States?

A Whether it has been physically accomplished or not I don't know. There are people certainly working on it.

Q Did I&E bulletin 7906 address that situation?

A I believe it may have.

Q All right; 7906, as I understand it, directed operators at plants that had coincident logic ECCS actuation to manually turn on the high pressure injection upon the pressure set point being released, being reached, regardless of the level set point.

Does that comport with your recollection?

A That does.

Q And 7906 went out to all licensees, didn't it?

A Some of those numbers went only to Westinghouse and some to Combustion. I could tell you in a minute.

Q Yes. Could we check on that?

A I have a copy of it right over here.

Q If you could check on 7906 as well.

A Yes.

(The witness checked the document.)

THE WITNESS: Went to all PWR's other than B&W .

BY MR. KANE:

Q All right. That is the direction to manually turn on the HPI when pressurizer level set point is reached regardless of whether or not the pressurizer level set point calls for automatic initiation of ECCS?

A That is right.

Q And 7906A only went to Westinghouse licensees, is that right?

A That is true.

Q And 7906A directs Westinghouse licensees to adjust level trip point such that ECCS actuates on pressure alone, is that right?

A That is true.

Q Is it only at Westinghouse plants that they have coincident logic ECCS actuation?

A That is my recollection. I haven't been involved in the detailed review of this aspect of the bulletins and orders, but it is my recollection that Combustion did not depend upon coincident logic.

Q So as far as you know, there is no coincident logic ECCS actuation at CE plants, for example?

A That's right. If there were, I think they would have been fixed by the same thing.

Q I was curious as to why 7906 talking manually about turning on HPI went to all licensees if you only had that situation of coincident logic ECCS at Westinghouse plants?

A Well, because all licensees had level indicators and for the problem that the level could stay up when you had a small break, they wanted it manually initiated.

Q I think that was addressed in 7905, as a matter of fact, under the specific circumstances.

A These bulletins ~~are~~ were coming out rather fast and furiously with technical staff kind of dispersed between here and Pennsylvania, and there was some repetition, and some lack of clarity in the initial bulletins.

Q I see. What is the rationale for coincident logic ECCS actuation? Why have that?

A Well, I guess--I don't know specifically what the historical reason was. It is couched in a much more difficult question than appears on the surface.

You have got a lot of engineering safety feature actuation signals and a lot of engineered safety features and you are protecting against a number of transients and accidents, and each design is different.

The ECCS high pressure ECCS for B&W, for example, is a full pressure ECCS. It goes all the way to full operating pressure and it utilizes pumps that are identical to the normal makeup pumps.

Westinghouse's design is much different than that. It does not use normal makeup pumps for ECCS; ~~and~~ it uses high pressure safety injection pumps which are not capable of full operating system pressure. None of them are in normal service, and in fact won't operate at high pressure, won't discharge anything if ~~under~~ the reactor coolant system pressure ~~that~~ is above their shutoff head.

Down through the years, the way those systems were ~~are~~ integrated and their overall performance for transients and accidents determined what their initiation signals would be.

I don't remember the specific reason, but it probably had something to do with the fact that the high head system~~s~~ on a Westinghouse design is of a lower shutoff pressure than it is on a B&W design.

Q If ECCS actuation were only tied to pressure in the reactor coolant system rather than to level in the pressurizer as well, how would that pose a problem for the high head injection pumps at Westinghouse plants?

A I don't know the answer to your question. I could sit down and go through the logic. It would take me sometime to do it.

Q In fact, if it is tied just to pressure, the pressure drops to a certain point, and the ECCS comes on?

A Yes.

Q High pressure injection comes on, presumably it brings the pressure back up in the primary system to the point where the high head pump would no longer function?

A Right, yes.

Q It would keep it up to that certain point, wouldn't it?

A Yes, but the logic must have had someth ng to do with protecting the reactor system from overfilling and overpressurization by too much high head delivery.

Q Wouldn't the situation also have to do with the voiding the situation of spurious or unnecessary ECCS actuation?

A Oh, yes. I am sure that that is part of the reason.

Q By tying ECCS actuation to two factors, level and pressure, rather than just to one, you are minimizing the number of unnecessary ECCS actuations?

A That is probably a factor.

Q And you are doing that as I understand it at the expense of having more of a hair trigger ECCS actuation system where you would be more likely to have an ECCS actuation?

A Yes, you are probably doing it when you are making that kind of argument from the traditional concentration on the larger breaks rather than the concentration on the possible intricacies of small breaks.

Q Okay. Isn't that also the situation, minimizing spurious activation of the system, the rationale for having containment isolation tied only to PSI in the containment building?

A . Yes.

Q In other words, if you tied it to ECCS actuation and high pressure injection actuation as well, you would have more containment isolations occurring?

A Not necessarily; you would have more, but not as often as your words imply.

For example, you might put it on receipt of the engineered safety feature actuation signal rather than upon initiation of the high pressure pumps, so for example, simple manual actuation of high pressure safety injection pumps would not necessarily isolate containment, but an automatic call for those high pressure pumps would, so it would be more frequent, yes.

Q Again then the tradeoff that is being made is between having more spurious or unnecessary actuations of containment isolation as opposed to tying containment isolation to a factor which will give you fewer such unnecessary actuations?

A That is true. Now in the case of high pressure injection into the primary system, there are safety reasons to want to keep the number of spurious actuations down to avoid putting ~~put~~ cold water into a hot system.

Q Puts strain on the system?

A Puts strain on the system; in the case of containment, it is not that obvious.

Q But obviously it also creates a situation, for example, in connection with ECCS actuation phenomenon not previously concentrated on at the time that the design was set up that can occur where the actuation will not take place when it is most needed?

A That's right.

Q And in talking about the ECCS, we talk about one of the most basic safety systems in a nuclear reactor, aren't we?

A Yes, we are.

Q We mentioned containment isolation being tied to 4 PSI. Are you aware that there was an estimate made in connection--let me ask you, are you familiar with the

Pebble Springs licensing proceedings?

A Yes.

Q Are you aware that in the context of those licensing procedings there was a question raised, question No. 26, that dealt with the system response to a transient which involves loss of auxiliary feedwater?

A Question No. 26 by whom?

Q I believe it was by the ACRS.

A By Mr. Ebersole I believe?

Q Right. It deals with loss of auxiliary feedwater.

A I don't recall that specific question. I can walk over here.

Q Can we pull that out?

A Yes, I think so.

Q The specific reason I am asking, Mr. Mattson, is that I am informed that the applicant's response to question No. 26 in the Pebble Springs licensing includes a time scenario of the system response that indicates that HPI would commence at about 10 minutes upon containment pressure activation signal 4 PSI, so I assume the containment isolation would occur within that timeframe.

At TMI 2, as you know, it took about four and a half hours for containment isolation to occur on that basis. The question arises then why is there such a large difference between what happened at TMI and what was predicted at Pebble Springs, which is a similar plant I am told.

A I have the question here in front of me. I have to admit I haven't read it in that light before. Let me look at it quickly.

Q Sure.

(The witness reviewed a document.)

THE WITNESS: The answer is probably something like the following--they did an analysis where all feedwater was lost, loss of main feedwater and loss of auxiliary feedwater.

Recall that at Three Mile, that auxiliary feedwater was turned back on at 8 minutes, so part of the answer is the extra 2 minut[illegible] of discharge at decay heat rate in the case of the Pebble Springs reactor would have given more discharge into containment, and hence higher pressure. That may be sufficient to get to the 4 PSI.

At Three Mile, it went up a couple of pounds and didn[illegible] go to 4. Another difference perhaps is that in doing a calculation of this sort, the models of discharge through the safety valves would have probably been conservative in the sense of overestimating the discharge rate from a safety valve. Both of those things would be inthe direction of an earlier receipt of the 4 PSI containment overpressure signal.

BY MR. KANE:

Q If I can paraphrase your answer, your suspicion is the consequences of a very small LOCA simply were not considered in connection with this analysis for the 10 minute actuation?

A Yes, were not considered in the realistic vein in which they probably should have been. One of the lessons from Three Mile is that there are some accidents where what is realistic and what is conservative may not be what is obvious on the surface from previous concentration on large break loss of coolant accidents. This is one of them.

Q I am curious as to what has been done by NRC in connection with technical fixes that may be incorporated into plants around the country, or specifically into TMI.

Are you familiar with the analysis performed by the NRC regarding the reactor system response to transients like TMI 2 when the NRC is considering technical fixes and do you know what computer codes are used in that regard?

A I believe we are using two. One is RELAP. I don't know which version of RELAP; I ~~suggest~~ suspect it is ~~from a suspected~~ RELAP 5, and we are using the code called the TRAC, RELAP being ~~a Rancho Seco~~ an INEL code, and TRAC being a Los Alamos code.

Q Would it be possible for us to get copies of those computer codes made available to our technical staff?

A Yes, I'm sure it is. They are available in code libraries I believe.

Q Do you think they are available in the public document room of the NRC?

A I will tell you the person to get them from the quickest is Dr. Tom Murley, Director of ~~the~~ Reactor Safety Research. The: are versions of RELAP running down the hall there.

A Has the NRC run the TMI 2 transient on a code like that?

A Yes--not the entire transient, the initial portions of it.

Q Who specifically did that?

A INEL and ~~[illegible]~~ Los Alamos are running ~~[illegible]~~ calculations. There is a document if you want to pause for a moment. There is an interesting summary written by a member of my staff as to all the calculations that are going off.

MR. KANE: Excellent. Let's go off the record for a moment.

(A discussion was held off the record.)

BY MR. KANE:

Q This document you are referring to references the results of running the TMI 2 transient on a code like this?

A In a summary fashion, yes.

Q Would Dr. Tom Murley have the detailed results?

A Yes.

Q We would like to arrange if we can to obtain those results for the technical staff of the Commission. Is there any problem with that do you think?

A You may run afoul of the fact that those calculations are being done for the NRC special inquiry and what their status is and what have you, they are sort of in an independent posture. I have specially sent you to Murley because the people they are using normally report to him, but they are currently reporting to the Rogovin people. I can't speak for what Murley can or cannot do in that regard.

The results of the B&W calculations using CRAFT are available. They have been submitted and are publicly out. They are the best I have seen so far. They get it to run longer and do better.

Q What other cases has the NRC analyzed concerning perturbations to TMI 2 which may include proposed fixes?

A I don't know of any that we have run. I do know that one of the requirements that has been placed upon the four venders is to run permutations and combinations of events of the sort normally analyzed in Chapter 15 of the safety analysis report, and whether the intent is to include permutations ~~or~~ and combinations of the TMI 2 sequence I don't know for the small break loss of coolant accident.

There have been several ~~perturbations~~ permutations and combinations of TMI 2 events already analyzed by Westinghouse, Combustion and B&W, and it is through those analyses that the operating guidelines and operating procedures for small break LOCA's are being implemented in operating plants today.

That sort of analysis, the person to talk to is ~~Sol~~ Zoltan Rosztoczy. He is ~~the Director,~~ the Chief of the Analysis Branch.

Q He is familiar with these other studies that are being done?

A The ones that are being required of the venders, yes. He probably is knowledgeable of what is being run by research since it is his field.

Q He would have those results to the extent they are available?

A Yes.

Q I am curious. There are several references in the lessons learned interim report that has come out, ~~new reg~~ NUREG 0578, to containment isolation and certain facets of that situation, but there is no estimate that I can find in there of type 2 containment isolation under certain circumstances.

Why not? Isn't that something that has come up in connection with TMI 2?

A Yes. That is one parameter that clearly is important, but it is highly variable. It depends upon the kind of accident you are talking about, the kind of transient with degraded auxiliary system performance. It varies very much.

I believe where you may be headed is shouldn't there be isolation on radiation, and I think that is treated in the discussion there. As we said, yes, we think maybe there should be, but we are not ready to require it.

Q I am more curious as to the calculations that were related to containment under various proposed types for containment isolation actuation under different types of circumstances of very small break LOCA's.

Is that going to be done?

A Probably not. I think that by tying it to engineered safety feature actuation, that you have clearly got containment isolation before you have got difficulty with the core.

Q As long as you also have--wait a minute. You were talking about tying it to the actuation of engineered safety features?

A Yes.

Q Which might or might not include high pressure injection?

A Would include. There is a way to defeat that. We were talking about ~~ACRS~~ ATWS. It is not in the design basis today, and it has historically been fought very hard to keep it out of the design basis by the industry, which is ATWS.

It may be possible to have a very damaging low probability anticipated transient without SCRAM that would lead to prompt failure of the primary coolant boundary prior to either high pressure in the containment or initiation of engineered safety features, and might be able to contrive enough things to find a way that you ought to use radiation.

Q Suppose you also have coincident logic ECCS actuation? Wouldn't that mean that you won't have this containment isolation occurring on the basis of activation of engineered safety features?

A Yes, could be.

Q Again that is something that needs to be changed and has been addressed in those I&E bulletins we talked about?

A Yes.

Q You mentioned before that you are somewhat familiar with the Pebble Springs licensing process. Are you familiar with question No. 6 that was drafted by Jesse Ebersole in connection with the Pebble Springs licensing?

A All of those questions were drafted by Jesse Ebersole. I have it in front of me.

Q Question No. 6 asking if the applicant knows that time dependent level will occur in the pressurizer steam generator and reactor vessel after a relatively small primary coolant break which causes coolant to approach or even partially uncover fuel pins.

Jesse Ebersole has already testified in the deposition we have taken that this question was prompted by the concerns that were communicated to him by Mr. Michelson.

The second part of the question asks whether or not, or what the operator proposes to do in light of those circumstances.

It is my understanding that the applicant's response the response by PG&E, does not really address the issue of time dependent pressurizer level indication or what they propose to instruct the operators to do under those circumstances.

The question is why doesn't it address that?

A ~~I~~ You will have to ask PG&E and the ACRS. They were there, and to my knowledge, the staff performed no technical review of those questions and answers.

Q If I asked you then what followup was done on this by the NRC, would your answer be you don't know?

A Oh, I know--none.

Q Are ACRS questions that bear on an applicant like this routinely routed to the Division of Safety Systems at the time that they are propounded by the ACRS to the applicant?

A Yes.

Q Routinely what does DSS do with those questions?

A Well, usually it works a little bit differently than that. The ACRS' subcommittee would have a particular interest in a particular plant and would convey those interests to the staff and the applicant, and the staff would assume that in the course of the licensing review of the facility, the documentation and the answer and the review of the answer and what have you was completed.

In this particular case, there is another anomaly in the usual communications between the ACRS and the staff; it seems that with the Michelson report there were several anomalies. This is another one. I recall these questions. They came to the attention of the Division of Systems Safety very late in the review of Pebble Springs.

Our review was done. We had issued an SER, Safety Evaluation Report, and within a matter of a very few days before the full ACRS meeting, the project manager for the Pebble Springs facility or his branch chief, I don't recall which, brought a copy of these questions to my office and told me that the chairman of the Pebble Springs subcommittee, Mr. Ebersole, had phrased these questions.

I looked at them. There were a large number. I didn't do a detailed review but I looked at them enough to ascertain that they were well beyond the kind of questions customarily asked in the course of a CP review under the Standard Review Plan. That is, they go well beyond the Commission's regulations.

In fact, if my staff were to ask me if they could ask those questions, I would say no in the main.

Q You wouldn't feel that DSS could even raise questions such as question No. 6?

A No. 6, maybe we could raise;No. 26 we could not-- failure of all feedwater is not a design question. Instead we would say demonstrate the reliability of the emergency feedwater is acceptable. That kind of question we could raise, but the kind of questions phrased in terms of assume you lose all feedwater despite what they have required you to design for and tell us what the consequences are is not allowable.

Q Why not?

A Because office practice is you stick to the Standard Review Plan and if you want to ask questions or generate requirements beyond that, you go to the ratchet committee and get approval to ask those questions and go beyond it.

Q In other words, the DSS is not free to pose whatever questions in the licensing process it feels relates to the legitimate safety concerns? It must stick to the SRP?

A That is true. I won't say it quite as strongly as you do because we do ask questions. We don't state requirements beyond the SRP, but we do ask questions, but fishing expeditions as large as one which says fail all feedwater and tell me the consequences are just too big a fishing expedition. The staff is not allowed to ask those kinds of questions.

Now the question No. 6 may have been a legitimate question in terms of standard operating procedure of the office, but let me go back to the story.

A large number of questions were brought to my office several days, less than a week is my recollection, before the full committee meeting, with an oral request as I understood it from the subcommittee chairman for the staff to develop answers and come prepared to speak to the full committee on these questions.

I stated that wasn't possible, that staff assigned to Pebble Springs had long ago finished their work. They were working on other things, that these questions in the main were beyond the pail of the Standard Review Plan, and that if answers were needed by the ACRS, they ought to go to the plant directly and get those answers.

That is what they did. I believe when the full committee meeting was held, that my staff was present and listened to the presentations. I don't know whether they were asked to comment on the answers that were given. I suspect to some extent they were, and when the committee, the ACRS wrote its letter approving the construction permit application for Pebble Springs, I am quite certain that the assumption was that the committee was satisfied with the answers they received.

Q Have you had an opportunity to read the answer to question No. 6 provided by the applicant?

A When NUREG 0560 was published and it was contained in there, yes.

Q In reading it over, did you find any portion of that answer that responds to the question about what they proposed to instruct the operator to do under these situations where they have a time dependent level in the pressurizer?

A I had earlier noted the same deficiency you have noted.

Q There was no response?

A Recognize that this is a construction permit application, and their answer may have been well, we have got plenty of time to do that before we go into operation.

Q They didn't give that response, did they?

A They did not. They may have under oral questioning at the committee meeting.

Q We don't know. You don't know.

A There is a transcript. I haven't read it.

Q Okay, but apart from that possibility, it appears there was no response to this question.

Who is responsible to follow up to make sure that an applicant responds adequately to ACRS questions, on a pending license application?

A The ACRS.

Q Does the NRC have any role to play in that regard?

A When they ask us to, yes.

Q When they ask you to?

A Yes. Usually early on they would pose areas of interest in the course of their review, and we would usually agree with those. I don't know of examples where we disagree, and so we would be following up on the same kind of things they are following up on, but for something anomalous like that, this set of questions propounded late in the review, the circumstances I described were the ones that were followed.

Q Doesn't the project manager for a given project have the overall responsibility to assure that the issuance of a CP or an OL for a project is justified based on compliance with all safety requirements and other concerns raised during the licensing process?

A Safety requirements, yes; concerns, no.

Q Safety concerns?

A Safety concerns of the NRC staff, yes.

Q How about safety concerns of the ACRS?

A . To the extent that the ACRS needs the staff to follow up for them, I'm sure we do.

Q If the ACRS poses a safety concern in a question to an applicant--

A And does not get an adequate answer, the ACRS has never been timid about saying so.

Q That isn't what my question was. My question was if the ACRS has propounded a safety related question to an applicant, it gets no response, the project manager is made aware of that situation, is the project manager entitled to sign off on an OL or CP for that applicant under those circumstances?

A I would suspect it has happened yes.

If it is a concern raised by the ACRS in their letter, that is a different breed of cat. Recognize that the ACRS is 10 or 15 fellows, all of them with different interests, not all of them alike, and questions raised by one are not necessarily supported by the majority, so it is the role of the project manager to see that the ACRS concerns are addressed.

When an individual member of the ACRS speaks or writes or raises issues, I think that that is stretching the role of the project manager a little bit.

Q These were 26 questions propounded by the ACRS, weren't they?

A No.

Q Oh. Now I wasn't aware of that. These 26 questions were not propounded by the ACRS in connection with the Pebble Springs licensing process?

A No. They were Jesse Ebersole's questions, but recognize ~~recognizing~~ that he was chairman of the subcommittee of the ACRS on Pebble Springs, so he had special stature in this review.

Q Were these questions posed to the applicant on behalf of the ACRS?

A I don't know the form of the transmittal letter or who signed it. I am sure the applicant thought of them as ACRS questions.

Q Did the NRC think of them as ACRS questions?

You seem to be trying to lay it on to Jesse Ebersole.

A I am not trying to lay it off on Jesse Ebersole, so you misunderstand what I am saying. The ACRS is a collegial body. It does not usually propound questions of an applicant.

Q But in this case it did?

A They hold meetings and individual members ask questions.

Q In this case they did?

A They reach collegial conclusions, and they write them as a collegial body. This is an anomalous situation. the chairman of the subcommittee very late in the review comes forward with a large number of questions. I suspect he did it as chairman of the subcommittee. They were communicated to the licensee, the license applicant, and he responded.

The staff did not, to my knowledge, offer a filing as to whether the answers were acceptable or unacceptable.

Q You say offer a filing?

A Yes--did not state a position, did not offer a judgment on whether the answers supplied by the applicant to these questions were acceptable.

The reason for that is fairly simple. Most of the questions went beyond the Commission's regulations.

Q And yet the staff is called upon ultimately, the head of the staff is called upon ultimately to issue the CP or OL, the licensing for which or in the context of which these questions are being raised, is that right?

A That's right.

Q And if I understand from a rather lengthy session I had with Harley Silver, the entire focus of the licensing process is safety, is it not?

A Yes, whether the filing and the documentation and the analysis conform with the Commission's regulations.

Q Which may or may not be the same thing as focus on safety?

A Let me take an extreme example, to try to illustrate the point.

If somebody comes to the FAA and says tomorrow that an earthquake at O'Hare Airport of the magnitude of 12 on the Richter scale would kill everybody on the runway, no matter what airplane they were in, and they didn't have time to get off the ground, would you expect the director of the FAA to close O'Hare Airport tomorrow? I think he would say no, that is beyond the realm of our requirements. If we should consider that, we ought to do it thoughtfully and we will go back and research the risk from that sort of thing and the cost benefit and all those other things, and we can decide whether our regulations ought to reach that event.

That is an extreme event I think, an absurd one, but between here and there is lots of gray matter.

Q Let's see if we can explore the gray matter a little. Let's say the situation is such that an application is being submitted for permission to construct O'Hare Airport, hasn't been done yet, and in the context of that application, questions are raised such as you just mentioned. Should the licensing, the permission to proceed with the construction of the airport go forward and the construction actually begin while that other question is still pending and is unresolved in connection with this specific project?

A Yes, but these questions could be asked of any plant ever reviewed.

Q Was your answer yes or what?

A I am not going to answer your question.

Q I won't press you on that one, Mr. Mattson, but it is analagous to the situation with a project coming up for CP approval. Nothing has even been done yet as I understand of breaking ground, and they are asking permission to construct a plant, and the ACRS poses a safety-related question. No response is given to that portion of that question, and nobody follows up.

I guess what I am trying to get into is shouldn't it be the responsibility of the project manager or someone within the NRC to follow up on this?

A If it were to be that responsibility, then the Commission would have to say as a matter of policy that it ~~were~~ was because the Commission fairly clearly in my judgment says to the staff "implement our regulations." Don't go beyond them. The entire focus of licensing review, the time I have been in it, in the last two and a half years, has been stick to the Standard Review Plan, the existing regulatory guides and the existing regulations.

If you want to go beyond them, take them to the Regulatory Requirements Review Committee and do ~~them~~ that divorced from individual licensing proceedings.

Q I see. If ACRS asks a question that goes beyond them, and gets no response, it is not the business of the NRC?

A I don't know what you mean by not the business.

Q Not the responsibility.

A No, no on that case.

Q Not your responsibility, okay.

A The ACRS would say as a collegial body having reviewed the information supplied in response to these questions, we are not satisfied with the response to question No. 6. We, the ACRS, advise the Commission as to something you ought to consider, eventhough it is beyond your regulations, and get a satisfactory answer to it, and we want to review it or are willing to trust it to the staff to conduct the review and reach the proper judgment which they take either of those alternatives, depending upon the instance, and they do it on every letter. Then the staff would follow it.

Q How often does the ACRS meet?

A Once a month.

Q For how long?

A Three days.

Q Three days once a month, I see; we have already taken Mr. Ebersole's deposition and asked him why this response to question No. 6 was not followed up. What he explained to us was that about the time the question was propounded, he had a number of personal problems in his family and he was paying very little attention to ACRS work as a result, and therefore he did not personally push it, that the general situation within the ACRS is something along the lines you described before. Everybody has their own interests.

This was a strong interest of his, and not of some of the other members of the ACRS, and therefore as a body, the issue was not pushed because he was not there individually to push it.

Given that situation, given the fact that ACRS meets as seldom as it does, once a month for three days or so, doesn't it make more sense for the agency that has day-to-day responsibility for licensing and supervision of the operating of reactors and the licensing of reactors in the country to have responsibility to follow up on ACRS questions to be sure that some adequate response is provided?

A Yes.

Q Okay.

A Even if the answer is, it is our judgment that this is not important, that it goes beyond the regulations, we don't intend to follow it through to a conclusion, it ought to at least be formally stated that that is the case, and I go back to what I said at the start of this series of questions.

I don't believe the staff formally said anything about these questions or the response offered to them.

Q As far as we can tell, as far as I know, this particular concern just kind of dropped into the cracks somewhere.

A Yes. There probably is no record of whether the staff thought some of these were good, bad, or indifferent responses.

Q Okay. We have been talking about this review of designs relating to plants.

Does your organization in any way review designs of systems that are not safety related on the secondary side of the plant or the balance of plant type of thing?

A Some, and the scope of the word safety related has been slowly increasing over the years.

Q Mind you I am focusing on pre-TMI 2 learning in that regard.

A We have been getting into control systems to a greater degree in the new designs than we did in the old designs, and ~~a~~ another good example is the emergency feedwater system.

In the early designs, it is not safety related. In ~~The~~ the Standard Review Plan~~s~~, it is safety grade. In certain cases, we have gotten more and more into what the word safety grade means for the emergency feedwater system.

Q The auxiliary feedwater system?

A Yes.

Q How long has the auxiliary feedwater system been considered a safety related system?

A Now there are two terms we are using interchangeably here, and I am not sure they are. Safety related and safety grade--safety related I suspect "aux" feedwater has been treated in that sense since the beginning. Safety grade requirements were not imposed, to my understanding, until the Standard Review Plan was issued.

Q Okay. Does your office on a regular basis look for interactions between safety related systems or components and non-safety related systems or components?

A Some.

Q When you say some, what do you mean?

A The Control system is not safety grade, and there is a requirement in the Standard Review Plan that interactions that could degrade or defeat safety systems not be allowed in the design of the control system. That is one example.

I can think of one specific example that is probably indicative of some thinking that has gone on in the last several years, as we get into the system interaction thing and that is the example I used several hours ago which was the air conditioning system for assuring that the environmental qualifications of instrumentation or controls that were necessary for ancilliary equipment, for safety equipment, couldn't go on the fritz and defeat safety systems when you needed them.

Q Is there any requirement in your office or anywhere within the NRC that there be notification of licensee changes to non-safety related hardware or procedures?

A Requirements are generally stated that the licensee must conduct a review and make a finding that his change is not an unreviewed safety question, and ~~he has~~ there are procedures that he is required to follow and document, and if he reaches a decision that it is not an unreviewed safety questions, his files have to say so. He doesn't have to tell us.

Q Would those files be subject to any NRC review?

A It is my understanding that they are audited by the Office of Inspection and Enforcement.

Q What they do is they look over these files to be sure some justification has been inserted therein by the licensee for the change?

A I suspect it is more than that, that they spotcheck to see whether they make any sense.

Q It is just a spotcheck?

A Yes.

Q If DDS, for example, is not regularly informed of those kinds of changes, how can the NRC in this way evaluate the effects of those changes in the non-safety related system upon safety-related systems?

A Well, DSS would not be routinely informed in any event because we don't have the direct responsibility for operating plants, but your question is equally germane for the Division of Operating Reactors, and the answer is that the system of regulation depends upon the judgment of the licensee.

Q That continues to be the situation today, does it not?

A Of course, and with the size of the agency that you have in front of you that you are investigating, that will continue to be the case. There is no human way possible to do it any differently with the people and resources assigned to licensing.

Q I want to ask you about that resources question because it has come up at several points that I&E, for example, does not have any I&E inspectors who are walking around looking for work to do. They have an awful lot of requirements they have to meet and a lot of things they have to check out in their plant inspections.

I have gotten the impression throughout the NRC that most people have got plenty to do in their jobs, and certainly it is not a situation of sitting around thinking of how they might dream up safety questions that don't come across their desk in some fashion or have some recommendations to relate.

However, in deposing Harvey Silver, the project manager for TMI 2, he pointed out the applicants have not been required in the past in the licensing process to submit any history of failure on even safety related equipment, a history of the device, how it has performed, the operating experience, and that the NRC in effect is left to learn of that itself through through LER program, I&E inspections, et cetera.

Why do it that way? Why not put the burden on the applicant to give you that information.

A Ask the Commission. They have had a proposal in front of them for some years.

Q Do you know if any action has been taken on that proposal?

A Meetings.

Q But nothing to implement it?

A No.

Q What is the objection to that propsal?

A Burdensome.

Q That it is a burden on the industry you mean?

A Yes.

Q Again, when talking about burden, we are talking about cost, aren't we?

A Yes.

Q So far the determination has been made to leave that burden with the NRC?

A Yes.

Q Is there any reason to think the NRC is in a better position than the industry to develop that kind of information and provide it?

A If the NRC does 'it,' there will be less of it.

Q Because the NRC doesn't have the manpower and resources to do as thorough a job as the industry could?

A That is why I state the conclusion I state, yes.

Q Okay. Your lessons learned report, NUREG 0578 has some interesting points. Item 2.1.1 on page 6 says, "Safety systems are called upon to work more often than previously expected."

Had anyone in the NRC prior to TMI 2 been responsible for tracking reactor operation experience in that or similar areas to see if the assumptions used in writing the regulations were valid based on operating experience?

A Not to my knowledge.

Q That would be to the extent it is a function at all, a function of I&E I guess?

A . No. We had started down that road curiously with high pressure safety injection systems and auxiliary feedwater systems in this division, and had discovered that the people in the probablistic assessments staff in the Office of Research had also started down that road.

We had contracts ourselves that were going to start this fiscal year, and I believe ~~them~~ they are already going, ~~&~~ the idea was to obtain failure rate data for these kinds of systems for real plants.

Q Do you know how far that has gone?

A I don't know its current status.

Q That was a program within I&E?

A No, no--research.

Q Let me ask you a broader question. We have been talking about the PORV.

A It is a very important question, something that ~~is~~ needed to have been done. It doesn't appear to me that you would have gained that much from it by starting years ago. It might have been better to start at a year ago or two years ago, but the body of data to be synthesized and put together is only now beginning to generate sufficient quantity statistically to be useful, in my judgment.

MR. KANE: Let's go off the record a moment.

(A discussion was held off the record, and the deposition was recessed at 1:10 p.m., to reconvene at 1:40 p.m. the same day.)

AFTERNOON SESSION

1:50 p.m.

MR. KANE: On the record.

Whereupon,

ROGER J. MATTSON

having been previously duly sworn, testified further as follows:

DIRECT EXAMINATION (Resumed)

BY MR. KANE:

Q Mr. Mattson, we have been discussiong off the record for just a few moments here Section 15.6.1, of the Standard Review Plan which is entitled, "Inadvertent Openi[illegible] of a PWR Pressurizer Safety/Relief Valve."

Is this the section of the Standard Review Plan that prescribes the review to be given during the licensing process to a PORV and its use in a pressurized water reactor?

A Yes

Q I note that this procedure is some five pages long, and it appears to call for some fairly detailed analysis and testing of the PORV.

Is this section followed as a regular practice in licensing plants that have PORV's?

A Yes.

Q I don't find anything in here, however, which talks about the situation of a PORV sticking open under circumstances where there is a loss of all feedwater.

A That's right.

Q Why isn't that situation considered?

A Well, it is a compounding of events. Perhaps the more germane thing would be to ~~let's~~ understand first the two events that are analyzed.

The loss of feedwater is analyzed as an anticipated operational occurrence, and the cladding is required to maintain its integrity for such an occurrence.

In that event, in the analysis of that event, the PORV is assumed to not open as it is designed to open because that conservatively bounds the pressure peak that occurs during that transient for loss of feedwater, and the analysis has never been done, never been required to be done, different I don't know if it has ever been done where the PORV opens as intended and it sticks open. Rather, the ~~It is a~~ failure that is ~~has been~~ postulated, is the failure to open.

The other accident that is analyzed is the inadvertent opening of a relief or safety valve. I don't know specificaly, but I wouldn't be surprised to learn that the valve that is opened in the case of the B&W machine is the safety valve because it is larger than the PORV and probably yields more severe thermal hydraulic consequences in the reactor.

Those are the two that have been treated. They have not been compounded. Obviously you could take the 30 events that are in Chapter 15 of the safety analysis report and add various combinations of them and create new design basis events.

The design basis events grew up over the years as a statement of an acceptable set of events for the design analysis and they have been subject to some probablistic assessment as probability techniques have matured over the past few years, and there has been a general feeling for a couple of years that we ought to reclassify some events, perhaps moving some of them out of the moderate frequency category because operating history shows that they don't happen that often, and perhaps move some others into the moderate frequency category.

TMI 2 tells us that we ought to create a new event, that is, a combination of loss of feedwater plus a stuck open valve, as an anticipated event or accident. I don't think we know yet. We didn't do them in the past.

Q Who verifies that this section has been complied with by the licensee?

A Reactor Systems Branch.

Q I see. How do you go about that?

A The licensee is required to submit in the safety analysis report information concerning the inadvertent opening of the PWR pressurizer safety relief valve, and to show through analysis that the acceptance criteria are met using codes, analysis codes, techniques that have been reviewed and found acceptable by the staff.

Q Do any of the analyses that are discussed in connection with this section relate to actuation of ECCS? Do they contemplate that?

A Well, yes, I'm sure they do.

Q Because I do also make note of the fact that in Part 3, under review procedures, page 15.6.1-3, there is the statement, "The sequence of events from initiation until stabilized condition is reached is reviewed to ascertain..." then there are a number of things listed.

One of them is the extent to which operator actions are required. Another is the operation of engineered safety system that is required.

A Yes.

Q Again, if I understand it correctly, however, the circumstances of a small break LOCA resulting from a PORV sticking open are not analyzed in connection with operator action or ECCS actuation, is that right?

A Not quite; what this says is that you review the analysis. You determine how much the engineered safety ~~systems~~ is required, that is, accounting for single failures and the like, that you have still got adequate ECCS capability, and you look at the extent to which operator actions are required; if action is required within the first 10 minutes, you would generally not allow it. You require the design to accomplish its function without taking credit for operator action for 10 minutes.

You wouldn't look at the question of are the procedures and is the training and those things associated with operator action adequate. That is the omission that you will find systematically throughout here.

Q There wouldn't be any concern then or any consideration in connection with this portion of the SRP of operator training to avoid going solid, for example?

A Not in this section, no.

Q Did TMI2 submit an analysis in connection with this section of the SRP?

A I'm sure they did. Well, I shouldn't speak so quickly. The stuck open valve has been a requirement for sometime. How far back exactly it goes I guess I can't say. With a little Research we can find out.

I suspect that they did analyze it, but notice acceptance criterion 1A, consequences of the transient are less severe than consequences of another transient that results in a decrease of reactor coolant inventory and has the same anticipated frequency classification.

It is possible that they were able to show by qualitative reasoning that this particular accident was less severe than another one analyzed in Chapter 15 of their SER, in which case they may not have done it in detail.

Q The bounding calculations approach?

A That is true.

Q As we know from Michelson's report, under the circumstances of a very small break LOCA, that reasoning may not be valid.

A If you would like to pause a minute, I have got a copy of Three Mile Island's SER back there on the table. We could look.

The supplement I have here doesn't have it.

Q We were talking before about the application of the Standard Review Plan to TMI 2. I got the impression from the overall conversation that TMI 2 was not required to comply with the provisions of this Standard Review Plan.

A That is right.

Q So insofar as these are requirements under the Standard Review Plant, isn't it conceivable they were not applied to TMI 2 at all?

A It is conceivable,but the list of required transients and accidents to be analyzed in Chapter 15 of the safety analysis report has been a pretty consistent list down through the years, and was previously required under a regulatory guide that didn't speak to the acceptance criteria, and the methods of review.

It just required that such an event be analyzed. I think this is one that goes back well before the Standard Review Plan.

Q Were there any problems with TMI 2's submission ofthe analysis called for by this section of the Standard Review Plan?

A I wasn't in this office when TMI was reviewed.

Q You are not aware as of today that there were any such problems?

A No.

Q Do you think it was probably in a supplement, however, to the SAR--I'm sorry, the SE?

A No. I just looked in the SER and I didn't find it in the SER, so it must be in a supplement if it is there.

Q Why would it come about that it would be in a supplement and be brought up later on in the process?

A Often the staff will issue an SER that has open items in it to be filled in at a later date, and the SER is taken to the ACRS for its review, and supplement 1 to the SER is usually the response to the ACRS review, and subsequent supplements address the open issued as they are closed.

Q I see. In your lessons learned task force short-term recommendations, of course, you do have a whole section on performance testing for PORV's, and that sounds like a recommendation that should be called for on the basis of what happened at TMI 2, but as I am sure you are aware, one of our concerns is why the situation was the way it was prior to March 28, 1979, so I would like you to if you can tell us what was wrong with the NRC system of review that it didn't recognize the large number of PORV failures that had occurred prior to TMI 2 and resulted in a similar recommendation of what is being made now.

A Well, there weren't a large number of failures. There were three.

Q You made reference to the fact that on the basis of frequency, PORV's sticking open was not an uncommon occurrence.

A Three times it stuck open out of 150 roughly that it was called upon to function, so it is a failure frequency of 1 in 50.

I am not sure that anyone had evaluated operating experience to the extent of characterizing the failure frequency of PORV's, that is, the failure to close, prior to TMI.

If they had, they certainly hadn't communicated it in the kind of terms that it was communicated after Three Mile Island, that is, one in 50 ~~has~~ was the likelihood of a small break loss of coolant accident, which is certainly a higher probability than the previously understood probability of a small break loss of coolant accident, that is the kind of probability estimated in the reactor safety study which are in the 10 to the minus third to 10 to the minus fourth per year range, so the failure of regulation there I guess is to one, analyze the failure frequency of such components, and two, to understand that failure frequency in the context of the overall regulatory basis as the basic assumptions about failure frequencies like a loss of coolant accident.

Q If I understand the situation correctly, prior to March 28, 1979, the NRC did not have any formal program for compiling the operating history of such devices as, for example, the PORV in terms of LER's having been submitted in the past, and that data then being used to try and track the performance history of these devices?

A People were making progress in that area. Over the last several years there has been more and more attention being paid to LER's and many alternative methods of analyzing operating history have been suggested, and several of them were under trial use.

I spoke earlier today about the research and DSS joint interest in researching operating experience for ECCS, and auxiliary feedwater systems. Those are typical of growing recognition of the need to make better use of operating data.

There was a program in the--off the record a minute.

(A discussion was held off the record.)

THE WITNESS: Back on the record. There was a program in MPA for accounting of the operating experience and some trending analysis. That was more of a straight bookkeeping rather than an engineering evaluation of operating experience, and those things, I think there were several yearly reports of that sort issued in the last couple of years, but again, a general recognition that that was not adequate.

ACRS ~~in~~formed ~~the~~ a subcommittee on LER's. ~~that~~ People were generally reasoning together to find a better way to evaluate operating experience.

Q Are you aware of any recommendations for upgrading or qualifying PORV's prior to March 28, 1979?

A Yes.

Q What efforts are you aware of in that regard?

A Yes.

Q What efforts are you aware of in that regard?

A In connection with ATWS that we talked about earlier, the staff of the Reactor Systems Branch, and in fact NUREG 0460, the latest report on ATWS ~~that was~~, that we issued here in DSS last spring or last winter, spoke to the need to do qualification testing of PORV's for slug flow, ~~and to~~ that is two phase ~~slug~~ flow, and solid water discharge from ~~on~~ PORV's.

Q Was anything done on the basis of that recommendation?

A The Office of Research was surveying and has completed a survey of the facilities in the world for conducting such tests. I believe there are facilities under construction in Germany, France and Japan. There is no such facility in the United States.

Q Is it anticipated that those recommendations will be implemented at those facilities?

A No. These are facilities to conduct the tests. These facilities are capable of generating a enough steam at high enough pressure to conduct such tests of valves. You don't conduct the test on the reactor. You take the valve and put it on a conventional boiler and blow down steam and water through the valve.

Q Was there any other activity in that regard concerning recommendations to upgrade qualified PORV's prior to March 28, 1979 besides ATWS?

A Probably. The one I would be aware of in a larger context is a general program for pump and valve operability standards, which was to be a three-party arrangement between the ASME, the NRC, and the industry, pump and valve operability standards have been developed over the course of the last few years and are about to be issued by the American National Standards Institute.

There has been some concern about how they ought to be phased in their implementation to allow for the development of test facilities because some of these valves and some of their flow conditions are unique and large, and the full test capabilities do not exist in this country just as in the instance here of safeties and PORV's, but the problem is generally much bigger than safeties and PORV's, and I am sure they were discussed in that context.

Q Okay. Are you familiar with regulatory guide 1.33? It concerns keeping as is blueprints current at nuclear power facilities.

A I am aware there is such a regulatory guide.

Q Did Metropolitan Edison agree to comply with that guide as a condition for the issuance of its CP or its OL?

A I don't know.

Q Is there any way we could find that out?

A That might be in the SER.

(The witness reviewed a document.)

THE WITNESS: They are required to state it in I believe Chapter 1 of their SAR and in Chapter 1 of our SER, I don't find any statement.

BY MR. KANE:

Q Is that something that the licensee is required to agree to as a condition of the issuance of a CP or OL?

A You mean to agree with it, to meet the regulatory guide?

Q Yes, in terms of keeping blueprints current.

A Well, this depends upon the age of the plant. Each regulatory guide has an implementation section in it, and that implementation tells you to which plant it is to be backfit, if it is to be backfit at all, and I don't recall whether this was one. The number you say was 1.33, right?

Q Yes. I think it is 1.33.

A - Revision ~~Reg~~ 1 wasn't issued until January 1 of 1977, and the copy doesn't tell me when revision ~~the reg~~ zero was issued. ~~initiated~~. Let me read here just a second. It appears not. It appears that that was only applied to people who docketed OL applications after September 1, 1977.

Q Only for OL applications docketed after that time?

A Yes. That is what it says, but that is rev. ~~reg~~ 1, and it doesn't indicate how rev. ~~reg~~ zero might have been applied. It may have had a reach that was earlier than that.

Q Again, what would be the thinking behind not applying a requirement like this for upcoming OL applications?

A This was OL applications, OL applications docketed after September 1, '77.

Q Right. What I meant was not OL applications, but I guess OL issuances coming up, what would be the rationale behind not requiring this kind of thing, the obligation to keep blueprints on the plant, as is blueprints in current condition?

A Well, the rationale in this particular case probably had something to do with the difficulty of generating those drawings after they were used as opposed to keeping copies of those drawings as they were used for future plants. In other words, for the older plants once you used them and stored them or destroyed them or whatever you did with them, it would be harder to go back and regenerate them than in the case where you knew in advance you were going to have to save them and keep them at the plant. You could plan a document storage facility and retrievability and that kind of thing. I suspect that was the kind of thinking.

In hindsight, having seen the utility of such documents at Three Mile Island, and having lived through not having them available when they were needed, we said, lessons learned has said that they ought to be available at the site.

Q Prior to March 28, 1979, wasn't it perceived that having those kinds o. documents in that kind of condition could be very necessary under certain circumstances?

A If people had thought very hard about what would be needed in the event of a major accident in a nuclear facility, then they would have had such thoughts.

Q I guess what you mean is that people didn't think that hard about it?

A People didn't believe it.

Q All right. There appears to be a total lack in what I understand of the licensing process, a total lack of involvement by either the Environmental Protection Agency or HEW particularly in the areas of worker protection and public health and welfare standards to be implemented.

One of the commissioners on the Presidential Commission is very interested in that area, and he really would like to know why the NRC does not more fully utilize the resources that could be available to them from the Environmental Protection Agency and HEW.

A HEW has no, well, let me think a minute. EPA, let's start where there has been a lot of involvement, EPA, of course, is in charge under its federal regulatory council authorities, that it inherited when it was formed, for worker protection standards; occupational health standards are set by the Environmental Protection Agency, not by NRC, and they have had an occupational health standard group working down through the years to revise the current standards.

EPA is also very much involved in the as low as practicable routine emission work, and pursuant to 10 CFR, Part 50 of our regulation, they were involved in review and comment on that.

Subsequently they put out their own regulation, 40 CFR 190, which is the 25 millirem per year per individual from all sources in the nuclear fuelcycle which is the overriding federal standard on routine emissions from a nuclear plant, so their involvement in the area of normal emissions has been very important, a big role.

In accidents, their role has been in the emergency preparedness area, and in being the lead agency for setting or issuing emergency preparedness guidelines, protective action criteria, general form and content of emergency procedure criteria for state governments and what have you.

It is Now under an interagency group that I can't say the name of, I am not working in that field today, but EPA I think has been a leader in the emergency protective action guideline field down through the years.

I think it is unfair to say they haven't been involved or ~~is~~ that the work is devoid of their involvement. Bill Rowe was a major actor in this arena down through the years.

Q What role does EPA play in the plant-by-plant licensing process?

A None. They have no legislative mandate for such a role. They are a standard setting organization under their FRC authority, and the general environmental criteria kind of authority.

Q There is an EIS which is prepared usually on a nuclear power plant project, isn't there?

A That is a NEPA environmental impact statement. That has some summary kind of information on the safety of the plant, but it has, really in terms of the accident impact made in the past, very high reliance on reactor safety study kind of thinking, and concentrated in its environmental effects on routine emissions, radioactive emissions.

MR. KANE: Off the record a minute.

(A discussion was held off the record.)

BY MR. KANE:

Q While off the record, you mentioned the involvement of EPA in water quality standards.

A The non-radioactive aspects of water--thermal heat, chemicals and other pollutants governed by the Water Control Act.

Q What about HEW? Do they have any involvement in the plant-by-plant licensing process?

A No involvement there, and very little in the past in the standards area. NIH was becoming interested under some prodding over the last couple of years in the epidemiological aspects of radiation standard setting, and of course, in the last year, President Carter has had a thorough-going interagency study of radiation standards and HEW has aspired to a leadership role in that work.

What the conclusions of that have been I don't know. Its conclusions were coming due about the same time as Three Mile Island happened.

Q I see. The reason I asked is that I am curious as to whether or not it wouldn't be an appropriate forum within the licensing process for CP or OL issuance to have a situation whereby HEW's concerns about worker protection or public health could be raised and analyzed in the context of a specific plant.

A Well, that would say that you are going to abandon uniform standards for workers across the country and go in and set individual standards in individual plants.

You can certainly do it that way. It would be very inefficient and I don't know why you would want to do it. I think you would be on a better basis, a health impact basis if you have acceptable standards and require every plant to be designed to meet that standard, and that is the case today. The role for HEW that I see, and I think there is a very strong, positive role they can play, I think it is about time that that element of government was brought into this question to integrate radiation health effects with other health effects.

EPA seems to have been singularly incapable of accomplishing that, and perhaps HEW can do that, and I think that is a very important role that needs to be served, but I would concentrate on the standard setting area and leave the case-by-case review of design measures to meet those standards up to one agency.

Q All right. What sort of retraining--

A For one thing. every time you set a standard in the occupational health area, you have to ask yourself what does it do to the offsite risk, and every time you take a step for offsite risk reduction, you have to ask what does it do for onsite risk increase, so a good example is if you are worried about failure probability of pressure vessels, that it might be too high, one thing you might do is more in-service inspection ~~for~~ of pressure vessels to decrease the risk to people offsite of unacceptable consequences from pressure vessel failure.

When you do that, you very dramatically increase the risk of the worker who has to do that, do the in-service inspection of the pressure vessel. You have got that tradeoff. That really has to be done by a person who understands both sides of the question, and it is hard to bring another agency into a particular case and say review the worker protection unless you also give them a full picture about the public protection question.

Q I see. You would have to integrate all of that then?

A Yes.

Q That kind of effort has not been followed up to this time by the NRC?

A Oh, sure.

Q In the licensing process?

A Of course.

Q I thought the point that you were making was that the standard provisions which had been promulgated by HEW and EPA were being followed in the plant-by-plant licensing process.

A They are.

Q But the kind of more specialized review you are talking about of looking at the overall worker protection program that is devised by an applicant for a specific plant has not been directly brought to the attention of HEW or EPA for this kind of review on a plant-by-plant basis?

A No, but it is done by NRC.

Q Internally?

A Yes.

Q I see. Okay.

A Without the involvement of HEW or EPA on the case-by-case level, only at the standards level, and not HEW in the past because they haven't been involved in radiation standards setting.

Q Okay. What sort of retraining was required of B&W operators right after TMI 2?

A It started very specific and was concentrated on the specific sequence of events of TMI 2. It grew fairly rapidly to include training for small break LOCA's in general, and as you read A0578, there is a program, a three-phase program, starting with small break LOCA's going to core uncovery, and then ending in reanalysis of the Chapter 15 events, all with a realistic perspective intended to improve and increase the capability of operators to intervene in accidents.

Q The more immediate situation right after March 28, 1979, as I understand it, there were hearings on whether or not to close all B&W plants, and in fact all B&W plants were temporarily closed during that period of time, weren't they?

A That's right.

Q And there was consideration at that time of the retraining of B&W operators at those plants, was there not?

A Not B&W operators--there is no such thing.

Q Operators at plants supplied by B&W.

A Yes.

Q What was the program that was devised at that time for the retraining of those operators?

A Well, initially it started out to be retraining on the TMI 2 sequence of events on the B&W simulator.

Q Was there an entire program set up that would call for simulator training?

A Yes.

Q Did it also call for retraining of the operators in the TMI 2 type of scenario?

A Yes.

Q Did it also call for oral and written examinations at the ending of that training program?

A I believe it did.

Q Was that program in fact carried out?

A I understand it has been, and it has been increased and broadened considerably.

Q Did the operators have to take written examinations?

A It is my understanding they did.

Q We had some conversation with Mr. Skovholt about this situation, and what he explained to us at the time was that there is a tremendous pressure that built up here because the utilities could not afford to wait for such a substantial period of time in keeping these plants closed while these operators were retrained, and that as a result, the determination was made that the operators would only take the oral examination immediately, and that further retraining would be sche...ied as could be conveniently accomplished.

A That is possible. I was not involved in the decisions on restart that came out of the bulletins and orders task force, but I know that Mr. Denton raised the passing score for those retraining exams, and I understood that had to do with the written--I am probably mistaken. Mr. Skovholt is in a better position to know than I am.

Q Mr. Mattson, do you favor a national training program for operators?

A Yes. There are some good aspects to that. I will reserve on a final judgment for the work of the task force. We are going to speak to that, I think, but maybe not in exactly those terms.

Q Do you think some type of uniformized training program is going to be addressed by the lessons learned task force?

A The need for one, yes.

Q In fact, at the present time, each utility is left to its own responsibility in developing training programs, is it not?

A Under very general guidance from NRC.

Q Right. Do you think that something like a national program is a real possibility, could be done?

A Yes. I think the principal attractiveness to that is that there are utilities that do a consistently better job at some things than other utilities do, and it would be useful to take advantage of the people who know how to do it better and get everybody up to the same standard.

It is a sort of best achievable technology approach that you can accomplish on a national level that you can't do with 100 different training organizations.

I am taking your word national to not mean federal necessarily.

Q No, but in any event, something like a college or a school that operators go to from around the country and obtain the training on a uniformized basis such that you know or can have reasonable assurance that operators at Rancho Seco have been trained the same as the operators at TMI 2, for example.

A Except you don't want them to all be exactly the same because every one of them has got a different machine and a different control room, so you still have to accomplish the transition from the central place back to the specific place.

Q As a matter of fact, that was something else that I wanted to ask you about at some point, and I guess this is as good as any.

Standardization of nuclear--as I understand it, there is some four or five different kinds of nuclear power plants at the present time in operation, and beyond that, every plant has unique features about it, particularly in terms of balance of plant aside from the primary system because there are so many AE's.

A There are no two plants in this country alike.

Q Do you think that is a good thing?

A No.

Q How has it come about that there has been so little standardization of nuclear power plants?

A The utility industry has not been convinced somehow that it is in their best interest. The vendors have responded fairly well over the last few years and have developed standardized plant options that make sense and could be used.

The response by the utility industry has been poor, disappointing, words like that. Even when there are standardized plants for them to buy, they make changes to the standardized designs when they purchase them and submit them as ostensibly standardized designs, but they are not really.

Q What are the problems with lack of standardization in nuclear power plants?

A Well, you have to start over in each plant and review it from the ground up over and over and over again, and instead of being able to concentrate resources by the designer and by the regulator on a particular standardized design, you have to spread your resources over half a dozen designs for a given vender, and you don't have the time and the opportunity to get into as much detail and have as much understanding about a design.

Q Has it also posed problems for I&E in terms of the differences between plants?

A I'm sure it does, in the same sense it does for license reviewers.

Q Make it more difficult to carry out the inspection and enforcement function, doesn't it?

A Yes.

Q Does it also lead to a situation where it is more difficult to see generic safety problems that may apply to more than one plant because ach plant is different?

A It makes it more difficult to see their solutions. You can see the problems. Conceiving of the solutions is much more difficult.

Q Given the fact then that some of the problems with lack of standardization relate to some basic safety type of concerns, why hasn't it been the position of the NRC that they won't grant a license to a utility which does not standardize at least to an acceptable degree its plant design?

A There has been growing enthusiasm for that thought over the last couple of years. I have espoused it myself. The reluctance appears to be an unwillingness to interject the government in that free enterprise choice, and lacking concrete, specific examples of why it is counter to safety to have a custom design when you have got 150 of them either in operation or under construction already, it is kind of hard to say they are unsafe, thou shalt only consider standard designs.

It seems to be difficult for the government to make that choice. I have recommended on the last couple of supposedly standardized design applications that they be rejected. They weren't standard plants. They shouldn't be reviewed as standard plants.

Q Were your recommendations followed?

A No.

Q Has anyone ever analyzed how effective the two-step CP and OL process for licensing really is? What I mean is doesn't issuance of a construction permit and compliance by the utility and following up the actually built plant virtually mandate issuance of an operating license?

A Yes, it does.

Q What is the function of the OL review as a separate component from the CP review? Everybody knows the utility is going to get the OL, don't they?

A That is not the function of the review. The function of the review is to ascertain that the design is up to snuff. whatever it was, and you make the changes that are required to meet the regulations.

Q Is it true that at the OL stage the hearings that are held only concern the intervenors' objections and they don't give any de novo consideration of the more detailed FSAR that is submitted?

A That is what the Atomic Energy Act says, yes.

Q Is that do you think a proper approach to the question of evaluating the FSAR which as I understand it is much more detailed than the PSAR?

A If you are going to have a two-step licensing process, it makes sense. That doesn't make sense today. It ought to be a one-step licensing process. There ought to be frozen designs. They ought to be standardized. They ought to review them at final design stage. It is not quite the same detail as the FSAR. It has clearly got restraint of trade difficulties with it, but it is clearly a solvable problem if there are going to be more nuclear power plants. The agency said that two years running in proposed legislation.

Q I wasn't aware of that. There was proposed legislation two years ago?

A Yes. The last two sessions of Congress have had licensing reform legislation by the Carter Administration largely drafted by the NRC which said essentially that.

Q Something else I wanted to ask you about, too, you are undoubtedly familiar with WASH 1400, otherwise known as the Rasmussen Report.

Are you aware that WASH 1400 stated that too much emphasis was being placed on large break LOCA analysis and that it really is necessary to look at more small break analysis?

A Yes.

Q One thing that has come to our attention, for example, in connection with the hydrogen question is the low capacity of hydrogen recombiners to deal with large amounts of hydrogen in the containment building.

For example, it is my understanding that the reason that low capacity was considered sufficient is that it was always anticipated there would be large break LOCA's which would generate not that much hydrogen in the containment building, and the recombiners would have the capacity to deal with that.

Is that right?

A No.

Q Why wasn't it considered that the recombiners would not have the capacity to deal with large amounts of hydrogen in the containment building?

A Because it was concluded that protection that was provided for the core, that is, the alternative cooling mechanisms for the core were sufficient to prevent the generation of hydrogen--small break, large break, any break.

Q So there wouldn't be any hydrogen in the containment building?

A Very small amounts generated during the loss of coolant accident.

Q Whether this is a small break ot large break?

A That's right. Small breaks are even better than large breaks. They still are.

Q Except under the TMI 2 situation?

A Except where you turn off the emergency core cooling system, and then there isn't any protection.

Q Did B&W supply to NRR on March 29, 1979 the day after the accident, an analysis of the hydrogen/oxygen situation?

A At TMI 2 post-accident?

Q Yes, or during accident, if you will.

A You mean as part of the accident management thing?

Q Exactly; work they had done on the situation at TMI 2 that would relate to the hydrogen/oxygen problem.

A Submit in writing?

Q Yes. In other words, transmitted in some fashion to someone in NRR.

A No, not that I saw.

Q You were in charge of a team that was working on the hydrogen/oxygen problem, were you not?

A Yes.

Q And the members of your team were getting their information from a number of sources, weren't they?

A Yes.

Q Do you know whether or not any members of your team contacted B&W in that regard?

A Oh, yes. We were on the phone with B&W quite a lot on Thursday.

Q Did they supply anything in writing relating to hydrogen/oxygen calculations to your team?

A On Thursday?

Q Yes.

A No, not to my knowledge.

Q Did they on Friday?

A Not to my knowledge.

Q Or Saturday or Sunday?

A Not to my knowledge.

Q Okay.

A Maybe late Sunday or early Monday is when the thing started to get a little more formal from B&W; by that time I had left Washington, so I am not sure quite what might have been provided. I have never seen anything in writing from B&W.

Q On March 30th or 31st, did you provide any specific data or information to the NRC Commissioners relating to the hydrogen/oxygen problem?

A On the 31st I did.

Q Do you have that data written down anywhere that we can take a look at it?

A No.

Q Was it just done verbally?

A Well, let me show you what I have given to you already and you tell me whether you want more. This is a question that was asked on April 26th ~~by~~ of Chairman Hendrie. That took some time to get answered, and you remember there were some harsh words back and forth.

This is question Roman Num~~b~~eral 4, provide definitive analysis on the hydrogen bubble problem that occurred at TMI. There is a short answer. It says other than what we did at the time of the accident, we haven't done much more, but here is everything we have.

Enclosure 1 is ~~that~~ a summary by Tom Murley of the Research staff involvement in the bubble thing--what did we know then, know now, who did we talk to, and what did they tell us, and there are a couple of attachments to his enclosure 1.

Enclosure 2 is a set of handwritten notes and some typewritten notes that I compiled from my files, from Bob Tedesco's files, from the file cabinet at the Command Center at Three Mile Island, and from the records from the Incidence Response Center, ~~that we were talking~~ I also assembled the ~~to~~ notes of Warren Minners, ~~We also talked to~~ Vic Stello, Darrel Eisenhut, and whoever, and they are in chronological order, and it says who said what to whom, when, when they wrote it down, ~~and~~ the other thing that is of interest is the transcript from Saturday, the 31st, of the Commission meeting, which Professor Pigford referred to I think when he was questioning me in my testimony on ~~the~~ June 1st, and it is an unfortunate transcript because it has got a lot of garbles in it. There is a lot of it that was lost, but that was the principal communication between the staff and the Commission on the explosion potential of the hydrogen bubble prior to the information being made public and the ensuing difficulty.

Q I believe you stated that this documentation that you have been referring to, which is about three, four inches in thickness, was transmitted to the Presidential Commission?

A Yes.

Q It is my recollection from talking to Professor Pigford that his problem is that he is not sure in all of this stack of information exactly which data was relied upon to communicate to the Commissioners your task force's conclusions relating to the oxygen/hydrogen problem, and specifically I wonder if it is possible to reconstruct what data was used to come to the computation as to the oxygen generation rate and what that conclusion was?

A Yes. Let me pause for a minute.

Q - Surely.

(The witness reviewed a document, and a discussion was held off the record.)

MR. KANE: Back on the record.

BY MR. KANE:

Q Mr. Mattson, before we took the break, what we were discussing was the request by at least one of the Presidential Commissioners, Thomas Pigford, that you be called upon to designate the specific computations used by your team during the time of the TMI 2 accident to come up with an oxygen generation rate in connection with the hydrogen/oxygen problem, and to specifically give the numbers that were utilized in that regard.

During the break you have had an opportunity to look over some materials and notes that you have. Have you been able to identify the specific numbers used by your team in coming to a computation on oxygen generation rate?

A Yes. First of all, let me straighten the record out on a couple of things. I didn't head a team doing calculations. I was either in charge or in a deputy position in charge of the NRR involvement in the Incidence Response Center, on the day of interest for the hydrogen explosion question, that is, Friday night, late, the 30th, and all day Saturday, the 31st.

In that capacity, I was responsible for the hydrogen bubble and for a number of other things associated with the NRR involvement in the TMI accident.

The point I want to make is the hydrogen bubble was just part of it. What I did in the discharge of this responsibility was to ask two groups of people to look at the question of the hydrogen combustion potential for the hydrogen that was in the reactor coolant system.

That question was asked of me at about 2 o'clock a.m., Saturday the 31st of March. It had been asked of others in the Response Center about four hours earlier, and various questions had been addressed and some preliminary work was started before the question was asked of me at 2 o'clock in the morning.

That early Saturday morning all we succeeded in turning on in the way or technical problem solving was to ask B&W to calculate the explosive force of ~~an~~ a stoichiometri mixture of hydrogen and oxygen in the reactor coolant system.

That is a very conservative first question to ask, but people wanted to bound the problem. B&W was asked that question sometime in the course of the morning, early Saturday morning. I am not sure when. I left at 3 o'clock in the morning. They answered at 5:30 that morning, and said that the result of a stoichiometric mixture exploding in the reactor coolant system, if it was 1,000 cubic feet at 1,000 PSI and 280 degrees Farenheit, would be a pressure in the reactor coolant system of 14,000 PSI.

When I returned that morning at 9 o'clock, I obtained briefings from the people who had worked earlier on a number of problems, including this one, and by 10:30, had a conversation with the Chairman where he re-emphasized the importance of obtaining an answer to this question in a timely way.

There was no answer when I came back to work that morning. I turned on two groups of people at that point.

The first group was headed by Robert Tedesco, assistant director for reactor safety in the Division of Systems Safety. He worked with his own staff; a Mr. Shapakar, Mr. Butler, Mr. Millstead; and they in turn worked with the Knolls Atomic Power Laboratory, and Knolls did some caluclations of the rate of generation of oxygen by radiolysis in the TMI 2 environment, and the staff did some caluclations.

I will tell you a little bit about what those numbers were, but I would rather do it in chronological sequence.

Q Please.

A The work of Millstead, Buttler and Shapakar and Tedesco is chronicled in a July 9, 1979 memorandum to me which I asked them to write, and I passed on to Mr. Gorenson of the President's Commission. It is a fairly good summary memo of what they calculated, by what methods, whe ,
in addition to another memo I supplied on July 2nd, written by these same people--I'm sorry. I did not supply it on July 2nd. The Chairman did. It is probably worth getting that straight on the record; so let me find it.

Q Was that Chairman Hendrie?

A Yes. Chairman Hendrie supplied to Mr. Kellney on July 5th, 1979 another memorandum, this one signed by Mr. Butler dated April 25th, which has good summary information on the radiolysis calculations supplied by the staff, so between this April 25th memorandum from Butler and the July 9 memorandum of Millstead, there is a good explanation of what the staff did and what they relied upon from KAPL, to supply answers on the hydrogen combustion question on the 31st, and then subsequently on the first.

The second group of people I turned to was ~~Sol~~ Saul Levine, Tom Murley, and Bob Budnitz from the NRC Office of Research, who in turn went to other people for analysis of the hydrogen combustion question.

They went principally to a Mr. Bob Ritzman at SAI. I can't say what SAI stands for--Systems Analysis, Incorporated, and to the Idaho National Engineering Laboratory. Those were the two sources that reported to me that they had consulted on this subject.

I subsequently have learned some months after the accident they (RES) also consulted with the Brookhaven National Laboratory.

Q Okay.

A Now what they obtained from Brookhaven, Ritzman, INEL, is quite well summarized in written memoranda in the stuff that we supplied in response to that question before I talked about it earlier. It is buried, but you can find it. It is in succinct summary fashion.

It is mid-morning, the 31st of March. I have turned on those two principal people, Tedesco and Levine. At 1:10 p.m. Saturday afternoon I got my first answer back. Prior to 1:10 p.m. Saturday, I did not know the answer to that question. I am not an expert in that field, and I attempted no calculations of my own.

Others had. The staff had not other than the ones I had turned on.

Q You say others had. Which others?

A I am not completely certain. I have tried to research the transcript of telephone conversations and Commission meetings to ascertain who the others were. I am not completely satisfied that I know.

The Chairman had done calculations of his own Friday night, and he told me that when he called to ask the question at 2 o'clock in the morning, Saturday morning. He said he had done calculations. He was concerned with the answers.

The Commission transcript of late morning Saturday morning alludes to 10 percent oxygen. Commissioner Gilinsky refers to that number. I have never heard that number before.

Q Did Chairman Hendrie tell you where he had gotten the data to do the computations he said he had done?

A No.

Q Did he tell you what the computations were, I mean the numbers?

A No. He said he was calculating the rate of radiolysis ~~x~~ and the generation of oxygen and if it were staying in solution, that is, it was not recombining with the hydrogen, then he was concerned that it could be rapidly accumulating, potentially leading to a combustible mixture.

I said I didn't know. It was a good question. I would have to ask some others.

At 1:10 p.m. I got the first answer to the question I asked, and the answer came from ~~Sal~~ Saul Levine. He told me that he had spoken to Bob Ritzman of SAI who was then estimating that there was 2 to 3 percent oxygen then present in the hydrogen bubble. He thought it could be 10 times higher than that, but he didn't believe it. He didn't think it would ignite until it reached 8 to 9 percent oxygen, and it wouldn't detonate until it was higher than 8 to 9 percent by a factor of 2 or 3. That information is contained in my notes.

Levine had also talked to Sid Cohen of INEL. Cohen's statement was there would be 5 percent oxygen in four to five days, that 900 degrees Farenheit was required for spontaneous detonation in a wet environment, and it would be expected to burn before it exploded.

At 2:06 that afternoon, Tedesco indicated that he ~~has~~ had talked to Westinghouse and to KAPL. They had not done calculations by that time, but they had some qualitative ~~quantitative~~ answers. The notes indicate that Westinghouse ~~believes~~ believed that the oxygen would stay in solution, and at the low temperatures that were then present in the reactor coolant system, the recombination of hydrogen and oxygen was not likely.

KAPL's initial response was that they could not preclude free oxygen in the bubble at that time. The oxygen/hydrogen generated by radiolysis would not be likely to recombine if there was boiling in the core.

My notes indicate Tedesco gave me that information shortly after he received it from Westinghouse and KAPL, so at 2 o'clock Saturday afternoon, I had an estimate that there was oxygen being generated from four independent sources, all with known credentials in this field. The estimate of how much oxygen varied, but all estimates said that there was considerable time, a matter of several days, before there was a potential combustible mixture in the reactor coolant system.

Q On that basis of thatinformation, what did you report to the NRC Commissioners?

A My monologue will take you right through it. It is interesting to note that at 2 o'clock Chairman Hendrie and Mr. Case were in a Saturday afternoon press conference.

They had not received this information from me prior to that press conference. They did not address this question in any detail in a press conference.

At 3:27 that afternoon, there was a recorded Commission meeting in the Incident Response Center, or nearby, of all five Commissioners. This is the garbled Commission meeting I told you earlier that the tape was a poor tape and there is much of the quantitative information that is missing.

I do recall discussing the numbers I had received from INEL and Ritzman through Levine and telling the Commission of the qualitative information supplied by Westinghouse and KAPL.

The bottom line of that conversation that is in the transcript was that there were several days required to reach the flammability limit, although there was oxygen being generated, and I expressed confidence that we were not underestimating the reactor coolant system explosion potential, that is, the estimate of two to three days before reaching the flammability limit was a conservative estimate.

Said another way, we told them it would be at least that long.

At 4 o'clock that afternoon after the Commission meeting, I talked to Tedesco who had just talked to KAPL and received the first quantitative information from KAPL. He had also been in conversation with a Mr. Irv Pinkel of NASA. The man he talked to at KAPL was Ernie Vernes.

KAPL had agreed that a conservative estimate of the hydrogen and oxygen generation rate by radiolysis calculated by the staff on Friday and Saturday of 28 to 39 cubic feet per day was acceptable, That is about 13 cubic feet of oxygen per day.

KAPL agreed that the hydrogen/oxygen concentration was now approaching flammability. This is Saturday afternoon at 4 o'clock. That is, there was about 5 percent mole ~~more~~ fraction of oxygen and 4.7 percent mole ~~more meoi~~ fraction of steam in the ~~a~~ hydrogen bubble.

They didn't suspect we would reach a detonation limit at those generation rates for ten days to two weeks in the future.

My notes indicate Tedesco's statement to me on the basis of what KAPL told him, that the oxygen probably won't recombine under the conditions in the TMI reactor coolant system. It will stay free. There might be some recombination due to gamma flux, and their estimate of the consequences of combustion were 20 percent over pressure in the reactor coolant system if burned at flammability limit, but they added that they had run a number of experiments in closed containers with steam, oxygen and hydrogen mixtures, and they had never seen spontaneous ignition at the 5 percent oxygen concentration.

This is after the Commission meetings. I do not recall passing this information to the Commission. I viewed it as confirmatory of what we had been saying to them about an hour earlier, that is, there were several days before we had a problem. The KAPL numbers were a bit more negative than the Ritzman and INEL numbers that I had received from Levine, but they were very positive on the no spontaneous combustion question.

Based on that information, we took no further action on the bubble that evening. At 6 o'clock, there are some notes from Tedesco where KAPL and he talked again. They talked about some new numbers based on the most recent measurements of the size of the bubble, 880 cubic feet at 875 PSI.

The results of their calculations were 5.8 percent mole ~~mool~~ fraction of oxygen, and a 4.3 percent mole fraction ~~mool~~ of steam, and according to the flammability limit curve that they had sent by telephone at 4 o'clock that afternoon, ~~that~~ was right at the burn threshold, but again KAPL's advice was they had never seen spontaneous ignition at the flammability limit, that being much lower than the detonation limit.

At 6:45, my notes indicate a call from Vince Noonan who is branch chief in the Division of Operating ~~Nuclear~~

Reactors in charge of engineering analysis. He would be the expert on what the explosion, it if were to occur, would do to the reactor coolant system. He had been in touch with a consultant by the name of Merriman who also did a stoichiometric burn calculation which yielded a 20,000 PSI overpressure in the reactor ~~to~~ coolant system.

Noonan also had been back in touch with B&W who had accounted for the effect of water vapor on the explosi~~ti~~on, and that had decreased the overpressure to 7,850 PSI. It is not an overpressure, a total pressure, and accounting for the effect of an enriched hydrogen environment, they were able to reduce the total pressure to 3,000 to 4,000 PSI.

That was fairly positive news. We know that the primary coolant system of PWR can take pressures, dynamic pressures in the range of 3,000 to 4,000 PSI as a result of ATWS analyses that had been done.

Noonan said that his estimate of the failure of the bolts in the reactor coolant system, the pressure vessel head would occur at about 12,000 PSI--I'm sorry--11,000 PSI. and it would ~~will~~ take 12,000 PSI to fail the head itself.

I didn't know it at the time, but there was--and it is obvious in notes I got from ~~in~~ the Incidence Response Center -- ~~of~~ a conversation with Jim Taylor of B&W at

11 o'clock Saturday night saying they felt that recombination was taking place, there was no oxygen being generated. That information did not get to me.

At 11:30 I had a conversation with ~~Sol~~ Saul Levine who updated the INEL estimate. They were now estimating 2 percent oxygen in reactor coolant system, but there was an anomaly. They were saying it was being generated at 1 percent a day, and given we were three days into the accident, that number didn't comport.

They were using data from the Cooper Nuclear Station startup test, which is a boiling water reactor, scaled it to the TMI 2 decay power level, estimated 12 days to reach a 6 percent oxygen level, again inconsistent with the 1 percent per day.

Ritzman was reported to be working with AVCO at that point with no further information from him.

I ought to go back and note ~~when~~ that Tadesco said he talked with Irv Pinkel of NASA. I can't figure out what NASA told him. It isn't in his notes, and no one seems to recall what NASA's role was. The name is mentioned but in no specific environment.

I left work Sunday morning at 3:00 a.m., returned at 9 o'clock. I had a meeting with Levine, Budnitz, Murley, in and out of the meeting were Commissioners Hendrie, Gilinsky and Kennedy. It was at about 9 o'clock that morning. The purpose was to reach an NRC. Béthesda staff judgment on the hydrogen explosion potential.

Everyone in the meeting cautioned that there was a lot of information coming from a lot of experts, and it didn't all agree, they didn't all agree with one another, but ~~that~~ the Chairman and I were leaving for the site. It was our best opportunity to give advice to Denton and Stello at the site, and the President was due to arrive there in two hours.

Recognizing the uncertainties in what we had heard, we agreed upon the following numbers. Murley has recorded them slightly different than me in his note to the file. I will read them both.

We agreed on the basis of talking to the experts that I have described that 5 percent oxygen was a realistic flammability limit; 11 percent oxygen was a realistic detonation limit, that there could be no spontaneous combustion below 900 degrees Farenheit, that the oxygen production rate was approximately 1 percent per day, and that the present oxygen concentration in the bubble was 5 percent.

Murley says we agreed on the three limits-- 5 percent for flammability, 11 percent for detonation, and 18 percent for combustion, and for the ~~love~~ life of me, I am not sure what those three things mean.

I know what the two limits mean. I don't know what the three mean~~s~~. Tom will have to answer that question.

Q Tom being again?

A Dr. Murley . The Chairman and I left for the site. We passed this information to Mr. Denton and Mr. Stello. Mr. Denton summarized it for the President in his briefing at about 11:00 a.m. that morning. Mr. Stello's initial reaction upon hearing the information was that it was wrong, and that we had miscalculated the radiolysis rate, and that in any event, the oxygen generated by radiolysis would recombine. He was convinced it would. ~~It~~ He couldn't prove it, didn't have an expert to back him, but he knew we were wrong.

Q What was Commissioner Hendrie's reaction?

A He was not involved in most of that briefing. I did it alone. The briefing with the President was finished about noon. Mr. Stello and I returned to the Command Center at the observation site across from the reactor. He got on a phone with General Electric and the Bettis Laboratory. I got back on the phone with the people in Bethesda, told them we have to have a better answer on recombination;by 3 o'clock that afternoon, as the record notes indicate. all but a few of the experts had agreed to ~~that~~ the conclusion that there was no net oxygen generation, and by the next morning, they had all agreed.

Now the specific role of KAPL in that is confused. They say it one way. We say it another way. I tend to rely upon what Mr. Millstead and others recall they understood KAPL to say.

I point it out because KAPL says it slightly differently in their documentation of what information they supplied, and somebody other than me will have to resolve those differences if they are important.

Those are the numbers we relied upon. The documents I have described in the course of the chronology say how the people calculated those numbers, and I think that should answer the question.

Q Okay. I do have one other question. At what point during that entire process did you recommend evacuation?

A Not at all through that process; I recommended no evacuation on the basis of the explosion potential of the hydrogen in the reactor coolant system. I wasn't convinced we had a problem.

We recommended to the Commission they had days before they had to make up their minds. It got a little more negative Sunday morning. For some reason, my six hours off changed it from 2 to 3 days until it was flammable to it was flammable now, but it didn't change the story that we still had many days until detonation and there had never been any observed spontaneous combustion at the flammability limit, so my advice consistently throughout the concern over the hydrogen combustion potential was to not evacuate.

There were Commissioners, not the Commission, but individual Commissioners who were concerned with the need to evacuate, but our advice to them was to not he concerned with evacuation because of the hydrogen explosion potential.

Q In fact, at some point during the accident you did recommend evacuation, didn't you?

A On Friday morning, the 30th.

Q What was the basis for that recommendation?

A Two bases--there had been observed that morning a 1200 milli-R plume reportedly caused by occasional venting from the vent header in the waste gas system, a header between the makeup tank and the waste gas decay tank caused by the transfer of gases from the makeup tank to the waste gas decay tank, said transfer having to occur periodically in order to maintain the makeup system, the venting occurring because of a relief valve that was popping at a pressure lower than its design pressure, that is, 80 PSI instead of 100 PSI.

The waste gas decay tank capacity was diminishing. I was told, as were the rest of the people in the Incident

Response Center in Bethesda, that there was one and one half hours of gas decay tank capacity left, at which point the discharge from the vent header would become continuous.

Had it become continuous, it would have beer a sizable increase in the 1200 MR plume. We were told that there was no alternative, because of that diminishing waste gas decay storage capacity, to depressurizing the primary coolant system and attempting to use the decay heat removal system.

We believed at that time, and no one has shown us to be wrong to date, that ~~that~~ use of the decay heat removal system would have uncovered the core by causing the, or that depressurization would have uncovered the core by causing the bubble to expand, and the core would have been severely damaged, with the significant possibility of melting it down.

Faced with these two alternatives, I recommended that they take advantage of the hour and a half they had before they were about to turn off the cooling of the core.

Q Why was depressurization at that point a necessary step?

A The licensee said that there was no alternative to depressurizing if they were unable to keep the makeup system running because they in essence had to open it to the atmosphere when they lost their waste gas storage capacity.

Now when I said the licensee said, you have to recognize that I was receiving that information through a telephone link that involved passing information through three or four or five, maybe even a half a dozen different people before it reached me, and you in your Commission have testimony that says, the [redacted] ~~the~~ information that I relied upon, and that the rest of the people in the Incident Response Center relied upon, was not good information.

What turned out was that in an hour and a half, they had found a way to control the makeup system without causing a continuous release, even in view of the fact that they had diminished storage capacity for waste gas, and the way they did that was to lower the water level in the makeup tank.

They may have known that all along, and our understanding that they were going to continuous release may have been faulty. If I understand the testimony that has been given to your Commission by the operators, that is probably the case. I didn't hear the testimony. I read it in the newspaper.

Q During a prior appearance that you did make before the President's Commission, Mr. Mattson, it was suggested by Commissioner Pigford that the NRC could have adequately analyzed the oxygen generation question using data already available to the NRC, and specifically data from ordinary conditions in operating boiling water reactors and pressurized water reactors regarding net oxygen production from radiolysis.

Have you had any opportunity to confirm if that is true?

A That is probably true.

Q Why wasn't that kind of data available or brought to the surface in the course of this chronology you have just given us of the hydrogen/oxygen treatment?

A Crisis.

Q Crisis? It simply wasn't realized that was available?

A A failure to use the information that should have been standard, and I can only explain that because of crisis.

Recall that I said that INEL used the data from the Cooper boiling water reactor startup test to get its oxygen generation rate by radiolysis. Why INEL didn't realize the way they defeat that oxygen generation is by putting hydrogen overpressure on boiling water reactors I don't know.

You recall I turned to Mr. Tedesco for the staff calculations, one of the staff's leading experts on boiling water reactors, who turned to people in his Containment Systems Branch who understand these things also.

The staff's familiarity with that information from boiling water reactor routine operations was either insufficient or the staff because of the crisis failed to extend that knowledge to the unique situation at Three Mile Island. The answer lies in there somewhere, and I am not certain I can attribute it.

I had no personal knowledge of how you control radiolysis in a boiling water reactor. If that is a fault, so be it. That is not an aspect of reactor operations with which I was familiar.

Q Do you know if you had used this data that was otherwise available within the NRC itself whether or not the results would have come out with more oxygen being generated than could actually have been the case at TMI 2?

A Maybe we are not communicating. Had we used what Professor Pigford suggests, we would have said what we eventually said--there was no oxygen being generated.

Q I see. You come to the conclusion which in fact--

A Which we eventually came to after there had been, unfortunately, misinformation put on the street, and a fair amount of panic ensuing from that misinformation.

Q I would like to go back to the question to what extent has the NRC requested B&W to do analyses concerning aspects of the TMI 2 transient? Has there been any of that, or has that all been NRC generated and carried out?

A We have required B&W to benchmark their computer code, CRAFT, against the information plots and data from the TMI 2 sequence.

Q Have those results been made available to the NRC?

A Yes, they were a condition of the restart of the B&W plants.

Q Does the NRC independently verify those results?

A Yes.

Q By using the same code?

A No--by using the code RELAP at Idaho.

Q I see. That was done as part of the startup procedure again from B&W plants?

A Some of that information was available in that timeframe. Whether all of it was I am not certain. Dr. Ross would be the one who would know the answer to that question.

Q Denny Ross?

A Yes.

Q I am curious on the question of the lessons learned plan, there was a minority view relating to recombiner capability for post-accident containment hydrogen control.

If I understand the situation, there are 46 operating plants that currently do not have a recombiner capability, and instead have opted to in the past handle the hydrogen problem and containment via venting, and the minority view in connection with lessons learned is that those 46 plants should be required to have a recombiner capability.

Why is that a minority view of the lessons learned rather than a majority view?

A There were only four of us that felt that way, out of 22.

Q You were one of them?

A Yes, and we felt it sincerely enough that we wanted to push it forward as a minority view. There were other recommendations of the task force that were not unanimous and many that had less or had more than four dissenting votes. We only required two thirds majority to pass a recommendation, but they were not seriously enough held for the people who favored them to push forward with any formality.

Q That means you had 18 people out of 22 who did not feel these 46 plants should have recombiners. Why not?

A They thought that the need for recombiners should be considered in context with the long-term work of the task force as to how the hydrogen design basis is to be changed, and if the hydrogen design basis, for example, ought to be changed to 50 percent metal water reaction, the recombiners aren't the answer.

Q Because there would be too much hydrogen for them to handle?

A Too much too fast.

Q Is there technology for recombiners with greater capacity?

A No, but there are other solutions to the problem.

Q Besides recombining?

A Yes.

Q What would be some other possible solutions?

A Inerting--that is what is done on all boiling water reactors.

Q Why hasn't that technology been pursued in PWR's?

A It was pursued in boiling water reactors because a very small percentage of metal water reaction takes you into the flammability range because they are very small containments, and in fact it takes you into the detonation range very quickly, and they have rather small pressure capability, whereas the large dry containments have a greater volume so they can take more metal water reaction and produce a smaller concentration of hydrogen, and they also have higher design pressures so they can take an explosion better.

Q If I understand the situation at present, 46 plants in the United States should they have a LOCA situation in which they have a substantial generation of hydrogen in the containment building,they are going to have two choices. They can vent it, or they can try at that point to acquire some recombiner capability and get one hooked up?

A They can detonate it.

Q That third choice would be highly undesirable, wouldn't it?

A Three Mile Island evidently detonated quite a lot with no adverse public consequences as a result of that detonation.

Q But that wasn't an intentional act?

A No.

Q Under those circumstances, you wouldn't recommend detonation, would you?

A I don't know how to do it intentionally. It happens unintentionally because of sparks.

A But even if you had the capability you wouldn't recommend that, would you, detonation of hydrogen in the containment building?

A I Might recommend burning it; one of the solutions might be to install internal flame recombiners of high capacity. Such designs might be possible.

Q What is being done in dealing with the hydrogen danger at these 46 operating plants?

A There isn't that much hydrogen danger. Three Mile Island said you can take a large amount of hydrogen generation, explode it,and not damage the containment. For the large dry containments, that is substantial protection.

It took a 50 percent metal water reaction, and did not damage the containment.

Q They did have a 28 PSI pressure spike?

A In a containment designed for 60 PSI, and that is the same way with all the rest of the large dry containments.

Q Have any calculations been done that guarantee you can't have a hydrogen burn under those circumstances that would exceed 60 PSI?

A Yes. That is an easy calculation. You just do the stoichiometric combustion of that much oxygen--I'm sorry--that much hydrogen in an oxygen rich environment, and calculate the overpressure.

The large dry containments will withstand it. The worry you have is not the containment. It is damaging equipment inside a containment that is not environmentally qualified for the burning environment. There is a worry. I don't mean to make it go entirely away. Recombiners don't solve it. A decision has to be reached on whether it is safe enough to do a better job of preventing large amounts of metal water reaction by training operators, by improving emergency systems, et cetera, or whether no matter what you do to improve the prevention of large amounts of metal water reaction, you still ought to design for it.

If you ought to design for ~~them~~ it, then you have got two alternatives. I don't think venting is an acceptable alternative. I didn't think so before Three Mile Island. I don't think so today.

Q Because venting necessarily would involve release of some radioactivity?

A It is purposeful release of radioactivity. I believe that ought to be controlled to as low as reasonable achievable levels.

Q I see a recommendation in here that plant operators demonstrate a capability to install recombiners.

A Yes.

Q Was that a majority recommendation in lessons learned?

A No. That is the minority view.

Q Does that recommendation include capability for a hydrogen recombiner on the basis of adequate shielding being available to deploy the recombiners?

A Yes. That is covered in another recommendation, a shielding review.

Q Does the FSAR review that is conducted by NRC include a detailed evaluation of the shielding system in the plants that are under review?

A I am having trouble with the word detailed. There is an evaluation of shielding provided for worker protection. That is a review done by the occupational health standard people or the occupational health people in another division, not mine, and there are requirements that where equipment would be needed for accident mitigation over the long term, that the design include consideration of maintenance of that equipment, and that should implicitly include shielding review for things outside of containment.

Having seen the capability of Three Mile Island Unit 2 in that regard, the task force wasn't satisfied with it and have called for a re-analysis of the shielding provisions for safety systems and non-safety systems in all plants.

Q What is your understanding of why--let me ask you, was it your observation that TMI 2 during the course of the accident at least initially did not have sufficient shielding capability to properly deploy the recombiners?

A Well, they may have thought they didn't need it. The design probably didn't contemplate an auxiliary building that was already severely contaminated by other sources, and the thinking at the time it was decided to put the shielding in there, as I understood it, was that they did not want to contribute to the access difficulties they already had in the auxiliary building. There were very few places they could get access to by putting recombiners in service, they were going to make it even tougher, so they ought to shield them before they did that.

They had some time, and they put the shielding together in quick order.

Q But they didn't have that shielding readily available? It had to be brought in?

A That is true.

Q If I understood what you were saying, isn't it the requirement of the FSAR that they have adequate shielding available?

A Not that I know; I am not certain.

Q Aren't hydrogen recombiners safety-related equipment?

A Yes.

Q So the shielding that is necessary to deploy them would also come under the rubric of necessary safety-related equipment, wouldn't it?

A Not if not required to maintain them, and you can get access to the other safety-related equipment; under the design basis, the answer would be no. Suppose The recombiner had a long pipe and ~~are~~ sat two miles out on an island in the river. You didn't need to be around it. If for some reason you weren't worried about atmospheric releases, you wouldn't shield it, right?

Q Yes, but that is not the situation, correct?

A Right.

Q Under the circumstances that we were faced with at TMI 2, it was necessary to bring in shielding from offsite in order to be able to use the recombiner at all?

A I tried to explain that. I think the reason, I am not certain, but I think the reason was that although they could operate without shielding, you couldn't get close to them, which meant you were denying access to a large portion of the anxiliary building by placing them in service.

If the auxiliary building was already contaminated from other sources of radiation, that meant you were making a bad situation even worse, so they decided to shield them. It may have been in the original design review, it was decided not to shield them because you had plenty of access to the other areas in the auxiliary building because they weren't expected to be contaminated.

Q Then the determination was made that adequate shielding would not be necessary because you could stay away from the recombiners?

A Right. I don't know how you answer the question of maintenance if maintenance were necessary in the post-accident environment.

Q Why would you want to stay away from the

combiners?

Wouldn't it be better to have the shielding available so that you don't have to stay away from them while they are in operation?

A It would seem so, yes.

Q Okay. Is the reliability of non-safety related systems that may require use of emergency or backup systems considered in the design review of a plant?

A Well, to a certain extent. We talked about that this morning when I said that the control system is required to be designed so that its failure does not challenge or diminish the safety system, but the degree to which that review has ever been conducted is limited. That is why ~~how the~~ B&W is now required to do a failure modes and effects analysis for the integrat~~ing~~ed control system.

Q That was another question I had. Prior to March 28, 1979, was failure mode and effect analysis activity utilized to ascertain which systems should have a high reliability or for which special quality controls should be specified?

A Well, yes and no; no design to my knowledge ever had all of its systems subjected to failure mode and effect to prioritize what systems should have what quality assurance controls. Rather by judgments over a period of years the definitions of safety system were derived. Those were required to meet the special requirements on QA and testability and be gold plated, as we call them.

Over the years it has been recognized that following that approach leaves the possibilility that you are subject to systems interaction that you wouldn't have contemplated by doing a review another way, and steps have been taken to try to decrease that risk.

Q Which organization in NRC does that type of work that you have been describing? Is that DSS?

A Probably three organizations--DSS, DOR, and Research, the Probabilistic Assessment Staff in the Office of Research, principally DSS.

Q That was my next question. Is there any office which gives more of an overall view to that question?

A DSS. Most of the failure modes and effects analyses have been confined to safety systems, Safety systems are not defined by failure modes and effects analysis, but reviewed by failure mode and effect analysis.

Q I think you have also told me in the course of this morning's testimony that DSS has very little to do with the operator training programs, is that right?

A Very little.

Q There is no evaluation of the adequacy of operator training programs in light of design of certain types of machinery within the plant, is there?

A None (before TMI-2). There is today, not as a formalized institutional way of working, but there is as a way of conducting the review by the bulletins and orders task force that is going on today.

Q That has been since March 28, 1979?

A Yes, sir.

Q Is there any intent in this regard to look at control room design in connection with operator training?

A Certainly.

Q Specifically what I have in mind in asking that question is something that has come up again and again relating to the ability of the TMI 2 operators on March 28, 1979 to have ascertained the PORV was stuck open.

One of the things that has been cited quite often is the quench tank pressure and level indicator, and I gather one of the problems with that was that the quench tank level and pressure indicator is on the back of the control panel at TMI 2, whereas according to the testimony we have, the primary focus and attention of the operators during the transient understandably enough was focused on the front of the panel.

Did anyone prior to March 28, 1979 consider that placing the quench tank level and pressurizer indicator on the back of the panel was an inappropriate location given the necessity for that information during that kind of transient?

A We don't review control rooms.

Q As far as you know, no one did?

A That's right.

Q Is there any post-TMI wisdom on that particular subject that indicates that you now will review control rooms?

A Yes.

Q Okay. Will control rooms be reviewed in their entirety or just something that you will carve out in that respect?

A We are going to have to look at the control rooms that are already in operation and the control rooms that are under construction and decide what ought to be done to change them, to provide display and diagnostic information to aid operators that is better than what is there today, but coupled with operators that are probably better qualified than the ones that are there today, so I see changes occurring on those two fronts over the next year or so that will have to be coordinated.

Q I notice that there was a reference that you made which was reported in the "Nucleonics Week" for July 19th, 1979, relating to reactor operations management, and specifically this was a presentation you gave to the ACRS concerning the work of the lessons learned task force, and you made referenced to the fact that there was a feeling that there should be a shift safety engineer which would in effect be the same as a senior reactor operator, but someone who had engineering capability. He would be on the normal engineering staff of the utility. He would be a person who could assume the hat of shift safety engineer during a transient.

Is this the current thinking of the lessons learned task force, that there should be such a person?

A 0578 recommended a shift technical adviser. We changed his name a little bit and said he ought to have engineering qualifications, ought to be on shift, and what other things he ought to do besides be on call for crises.

Q When you say engineering qualifications, do you mean a bachelor degree in engineering?

A Yes.

Q And do you mean any further formal training beyond that?

A Yes.

Q What kind of formal training?

A Design and layout of the plant, capabilities and limitations of what is in the control room, specific training in the plant response characteristics and analyses that have been done of the plant response for off-normal conditions.

Q And this person during a non-transient situation would you, you envision to be a senior reactor operator?

A No.

Q What position would he normally occupy?

A An engineering position in the operations organization, dedicated to safety considerations of both a short term and long term nature, long-term things like maintenance policy, equipment procurement policy, what have you, from the safety engineer point of view, and in a short-term basis, the review of operating experience from his plant and plants of like design to understand the implications of that operating experience for his design and for his operations organization, whether his operators and senior reactor operators and his procedures ought to be retrained and rewritten in light of operating experience.

He, for example, would have been the person at Three Mile Island who would have reviewed the Davis-Besse transient, would have understood its import, would have seen that there was better training for his operators, and at the time of the accident, he and one of his peers would have been on shift.

Q On the other hand, do you mean really in light of everything we have seen about how the Davis-Besse transient was evaluated both within the NRC and I guess elsewhere, for example, Mr. Michelson stated he had heard of the Davis-Besse transient, do you really think that the shift safety engineer at a nuclear power plant would have had any greater success in learning the lessons of Davis-Besse prior to TMI 2 than NRC itself did?

A Probably not prior to Three Mile Island, but since Three Mile Island he certainly will have a new perspective.

Q Again, doesn't that suggest that a shift safety engineer is not going to be in a better position than DSS is to day to be able to learn the lessons of new and unusual phenomenon that occur in terms or pressurizer level performance?

A Are you a pessimist or an optimist? If you are a pessimist, you throw in the towel now and say shut them down. If you are an optimist, you say we can learn from the mistakes, and I think you can learn.

Q Again, it depends on what mind set you happen to have when you are viewing the problem?

A Prior to the accident at Three Mile Island, the mind set was that accidents wouldn't happen. The mind set after Three Mile Island has to be that accidents have happened.

Q I have had that explained to me in a curious sort of way. Let me see if you agree with this. I forget which of the many deponents I have spoken to in the last two weeks who brought this point out, but someone brought up the point that the original beginnings of design analysis for reactor plants was that accidents will happen and we need to design against them, so the original idea was that we must design to prevent accidents. The entire thrust of design analysis was to prevent accidents.

That eventually led to the point where it was built into the analysis that accidents would not happen because of all the previous design analysis that had already taken place, and you built up to a point where it was simply anticipated that certain types of accidents particularly could not and would not happen given the prior history of design analysis and performance I suppose to that extent.

One specific example that was cited to me was the fact that in the original design analysis, a large break LOCA was, of course, anticipated in the containment building, and that it was therefore designed to withstand the consequences of that type of accident.

However, as time went on, it was not anticipated that you would have a situation where the LOCA accident would be of sufficiently small proportions that you would not have, for example, containment isolation on 4 PSI, and that you would have, as you did at TMI, an automatic sump pump arrangement which would pull water out from the sump that was running off into the auxiliary building, and as you said before, contrary to people's expectations, severely contaminate the auxiliary building.

A That isn't what did it.

Q It was my understanding that what happened was the tanks overflowed.

A That isn't where the radiation came from. That is where water came from, but it was uncontaminated water.

Q ~~How did it flood~~ What contaminated the auxiliary building?

A The vent header in the makeup system, but the water was all over the auxiliary building, and once the contamination came out, it got spread all over the building.

Q That is the current school of thinking? In any event, the point that was made was that having originally designed against the accident, you reached a point where you no longer take those accidents into consideration in further design and improvements or changes.

Do you agree with that analysis of the history of the design analysis?

A No.

Q You don't think that has been the case?

A No.

Q You don't think the reason that it was simply not anticipated that certain accidents would happen was because originally those accidents were designed against?

A No.

Q Why has it become the situation then that certain types of accidents are simply not anticipated or designed against today?

A I think it is because the people mouth the word to a certain extent that these machines were designed to not have accidents.

If these machines were designed to not have accidents, the PORV would have been analyzed for failure to close. People would have understood the consequences of the PORV staying opening during a transient ~~upon~~ in which ~~during which it~~ safety was dependent upon ~~that transient~~ its closing.

transcription garbled

If they really believed they designed to prevent accidents, they would have qualified that valve before it was procured for that situation. They did not. Instead people believed evidently in providing safety systems, well engineered, well designed, well analyzed safety systems, and the fault was they believed so much on the infallibility of those safety systems, they forgot about the people who could stand by and defeat them if they didn't have the right training.

They forgot about the excellence that was needed at all phases of the operation and the design.

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See, the PORV didn't stick during a transient ? upon which it was dependent. The ECCS did a pretty good job on that. They worked heck out of that for 10 years. It worked pretty well. The operator didn't work very well. Why? Because he wasn't prepared to work very well, not because he was dumb or because he was incapable compared to other operators.

He was in a situation that nobody told him how the thing would look when it got into that situation.

Q Wasn't he in fact trained to take certain action which was, under the circumstances he was faced with at that time, diametrically opposed to the right thing?

A No. He was trained to protect against going water solid for a system without a hole in it.

A By throttling back the HPI?

A For a system without a hole in it--this system had a hole in it. He had a lot of indications that it had a hole in it. The combination of the training and procedures and ~~regs~~ requirements for that combination of events failed him.

Q Was he trained to anticipate a hole in the system when he had a rising pressurizer level?

A No.

Q So in other words, under those circumstances, his pressurizer level told him he couldn't have a hole in the system, didn't it?

A I think you have to separate it into two stages. Early in the transient, there was a lot of, he had to make some tough choices very fast--very trying. The transient went on for 16 hours.

Q Okay. Early he should have caught on.

A There were a lot of people with a lot of expertise.

Q Hasn't it been estimated that most of the core damage took place during the first three hours?

A There certainly was a lot of damage caused early in the accident, but the core stayed partially uncovered for a long, long period of time. It was over 2,000 degrees evidently between 8 o'clock and 9 o'clock. There was severe damage caused by that happening. How long it stayed up there, I don't think we know.

Q But to come back to my question, hasn't it been estimated that most of the core damage was done during the first three hours? I am really trying to recall if that is correct.

A Some people have said that, yes.

Q You have heard that statement made?

A From EPRI, yes.

Q Okay. Fine. EPRI is good enough for me.

A Not for me.

Q You mentioned the fact that the operator had to make some choices pretty fast in the course of that transient.

Why did he have to act so quickly under the circumstances of that transient?

A Might be because his initial reactions were wrong and it was also the need for ~~front~~ fast action was exacerbated by the failure to have those valves open.

Q Is there anything about the B&W design that calls upon the operator to act very quickly in the event of a loss of feedwater?

A Certainly.

Q What is that?

A We have spoken to the ~~sense~~ sensitivity that -- we as an agency have spoken to the sensitivity of the design. It dries out faster, the fact thatits volume swings on secondary side upsets are larger than the pressure and volume swings from oth PWRS. It is well documented in the B&W shutdown orders.

Q All of that was pretty well documented prior to the B&W shutdown, wasn't it?

A Do you mean prior to the accident?

Q Yes.

A Was that knowledge available, yes, yes, it was available.

Q Was known prior to the accident that the boilout rate of a once through steam generator under the circumstances of loss of all feedwater was about two minutes?

A That number certainly is amenable to calculation, yes.

Q And was it known generally prior to the accident that the same circumstances, the boilout time for the recirculation steam generator was as much as 30 minutes?

A Yes.

Q Was it recognized then within the NRC prior to March 28, 1979 that under those circumstances, the operator would be called upon to respond much more quickly by a very significant magnitude in the circumstances of a loss of all feedwater under the B&W design as opposed to a Westinghouse design, for example?

A Yes.

Q How could it come about that a system that places so much more reliance on the operator, that requires so much quicker response time, could still be considered sufficiently safe by the NRC for purposes of licensing?

Why have a once through steam generator? Why allow it?

A Well, you have to look at historical precedent because it certainly plays a role in that question. The sensitivity when they were first licensed was probably not understood in that same light, and once you licensed a couple, you started requiring more and more reliable feedwater systems, and at that point they ~~haven't~~ hadn't generated all that much operating experience to establish definitively what you say is true.

It is still quite possible the once through steam generator is completely acceptable for the future with a reliable system and reliable control rooms, and sufficiently good operator tra[illegible]g, operator qualifications.

Q Well, you know, I still keep coming back to that it sounds like what you are talking about is training the operators up to the requirements of the once through steam generators instead of opting for a design of steam generator which does not place those kinds of demands on the operators to begin with.

A Don't you support there were similar differences between the 707 and the 747?

Q I really don't know.

A Of course there were.

Q All I am asking is why. You can think of lots of reasons why the once through steam generator is not a good idea. I have difficulty thinking of the reasons why it is a good idea, and that is what I am asking.

Why have a once through steam generator? What is the justification?

A Well, they are--you can make justifications.

Q As I understand the thermal efficiency of a once through steam generator as opposed to recirculation is on the order of 1 or 2 percent greater. Is that right?

A Must bc something on that order.

Q So it is a very marginal improvement in terms of the output. Are there other financial considerations?

A Not that I am aware of.

Q There is one. Now that I think of that, I think it was explained that apprently the once through steam of a steam generator producing superheated steam--but it is extremely dry steam?

A That has to do with its efficiency.

Q And it does not require a further device which the recirculation steam generator requires?

A That is just the economy. That is the same reason as your first reason.

Q We are back to cost. Is there anything else besides cost that makes the once through steam generator better than the recirculation steam generator?

A Not to my knowledge.

Q We have got a situation then where again to be as simple as possible, whereas one device which is less efficient gives the operator as much as 30 minutes to react in the event of a loss of all feedwater, and another device which as I understand it even with the improvements that have been made now, certain adjustments that have been made since March 28, 1979, gives the operator as much as 5 minutes in the event of loss of all feedwater?

A No, no. You misunderstand. You can lose all feedwater indefinitely on the B&W machine because of its significant capability in the emergency core cooling system. If he leaves the emergency core cooling system on, he needn't use the auxiliary feedwater system.

Those analyses also show if you leave the emergency core cooling system and the auxiliary feedwater system both off for 20 minutes, you don't get into difficulty, as long as you turn one of them on at the end of 20 minutes.

Don't make it worse than it is. They boil dry in 5 minutes. They won't go to hell in 5 minutes.

Q What I am referring to is the consequent emphasis that this situation leads to in the operating procedures for the operator to do something once he loses all feedwater.

I have seen the emergency operating procedures for TMI 2, for example, and I have been amazed at the amount of the extent they are called upon to perform within the first 30 seconds, the first minute.

Now perhaps operators can be trained to do that. What I am wondering is why even consider the necessity of training them to do that if the design doesn't need to be set up that way?

A Well, you are suggesting a sort of best achievable technology, and that is not the system of regulation we have had. The words say no undue risk.

Q Okay. Why isn't it considered undue risk when you could give the operator up to 30 minutes in reaction time and instead you opt for a design that gives him much less in terms of normal attempts to bring the plant back to operating condition? Why have that situation? Isn't that an undue risk? It is not necessary to incur it, right?

A It is a different risk.

Q Okay. It is an unnecessary risk, isn't it?

A Depends on whether you are a B&W owner or not.

Q Are you aware of any reason why B&W couldn't adopt a recirculation steam generator in its design?

A No, but I haven't really studied the question.

Q Okay. Maybe there is some reason. Maybe it is patented or something, but I am not aware of any and it comes back to the question of man-machine interface, doesn't it? Once through generator works fine if the operator is well trained enough to work with it, but it can pose this situation.

What if you put a passive feedwater system on a once through steam generator, a big accumulator. I suppose that would solve the boilout problem, would it?

A Yes.

Q And give much longer boilout time?

A Yes, but also drastically increase the propensity to overfill the secondary side, which also has difficulties.

Q It doesn't pose the same difficulty of saturation in the primary system and uncovering the core, though, does it?

A Not unless it breaks the secondary system and you have got an uncontrolled discharge from primary and secondary, and then you haven't got anything.

Q The reason I ask these questions is because it comes back to the best illustration of it in the TMI 2 incident. Eight minutes into the event the operators discovered that the valves for the auxiliary feedwater system were closed, and they opened them and they got this auxiliary feedback, as I understand it.

If they had a recirculation steam generator with a 30 minute boilout time, they would have made it, wouldn't they? It would have gotten the auxiliary feedback?

A Not if it sat there for 16 hours with the PORV open.

Q They would have gotten it before they had a boil dry in that situation, right, in the steam generator, and therefore isn't it likely they wouldn't have challenged the PORV to begin with?

A No.

Q Hasn't that been the experience in the Westinghouse plant, that they have had almost no challenges to the PORV's compared to B&W plants?

A They don't use them for the maneuvering mechanism to the same extent that B&W does with the integrated control system. The PORV doesn't open because the steam generator boils dry.

PORV's have opened in B&W plants a total of 150 times because they are used as a maneuvering device in conjunction with the integrated control system.

Q What do you mean by maneuvering device?

A To avoid SCRAMming the reactor upon ~~SCRAM~~ loss of the secondary side.

Q This is another aspect of it, isn't it? The B&W design philosophy up until now has been to not have an anticipatory reactor SCRAM upon a turbine trip?

The Westinghouse philosophy has been to have the anticipatory reactor SCRAM upon turbine trip?

A Hence fewer challenges of the PORV.

Q And more reactor SCRAMs?

A That's right.

Q Okay. It is obvious from the TMI 2 situation and the steps that have been taken since then that presumable the NRC now recognizes that the Westinghouse philosophy in this particular area is more desirable from a safety point of view?

A I think so.

Q Okay. Again, the question becomes why was that fact not appreciated prior to March 28, 1979? Again wasn't it clear that that type of situation, no anticipatory trip, placed the operator in a situation where he could have his steam generator boil dry and he could have his reactor not trip for a certain period of time after his turbine did trip as a result of the boildry situation, and therefore again be faced with a situation where his PORV is going to be challenged and get a break in the preliminary system?

A It wasn't put together that way by anyone I know.

Q Okay. All right. Just so I can leave that subject, what is your best explanation for why it wasn't put together that way?

A I am not sure why the designer didn't put it together that way. The staff had never reviewed the integrated control system, the PORV, or the operating history with bad PORV performance, and had never systematical put all those things together, hadn't studied them individually, let alone put them together systematically.

Q Let me ask you this. The OL was issued for TMI 2 in February 8, 1978. If in connection with that OL issuance say toward the middle or say the end of 1977 B&W was, had been required to submit a history of the operation of the PORV in its plants of the once through steam geneator and of its boilout rate and things of that nature, complete documentation of the history of these various devices as they had previously performed in B&W plants, wouldn't that have indicated that something should be done about this design, based on what you know today?

A The OL review was finished significantly in advance of the time the OL was finished.

Q Six months.

A More than a year.

Q A year before?

A Oh, yes. What is the date on the SER?

Q September of '76, something like that.

A That is when the OL review was done except for picking up a few loose ends, two years before.

Q Doesn't that create a situation where you have that kind of time lag between the safety analysis or safety evaluation and the actual issuance of the OL that further events have occurred, the knowledge and data have become available that indicate something should be done in connection with that OL issuance for the conditions, for the requirement, something like that?

A Yes. I agree with that.

Q And yet they are not incorporated because of safety evaluation, something done earlier? Why is the SER prepared so much in advance of the OL issuance?

A We tried to gear the SER to be finished at the time they are ready to operate, and you do the best you can to review everything you can get in there between now and whenever that end date is.

Q Why isn't it possible to require the applicant to submit a supplemental FSAR to cover that time lag period between the SER and OL issuance?

A For long delays, that has become the standard practice. Salem 2, for example, was significantly delayed beyond Salem 1, and the staff did another review, with Salem 2 not yet complete. As a matter of fact, Salem 2 is ready to operate, and that review is not done.

Q When you say for long time lags, we are talking about the difference between September, 1976 and February, 1978. That is about 14, 15 months. That would not qualify as one of those long periods of time?

A No, still wouldn't, not today, not under current practice. I am not arguing with your proposal. That isn't where I would start. It is a decent proposal.

Q Where would you start in terms of the licensing process?

A Something like all OL's to be issued after January 1, 1990 shall have a full demonstration of conformance to the Standard Review Plan and reasons why not, plus anything that has been added to the plant as a result of Three Mile Island.

Q You would say that regardless of when the facility obtained its construction permit?

A That's right.

Q So that could lead to a situation in which the utility has invested millions of dollars building a plant according to the construction permit obtained in 1970, and now comes along and submits its application, has the bad luck I guess from their point of view to submit their application after January 1, 1980. and you require them to backfit all that stuff from the Standard Review Plan or otherwise justify it?

A That's right.

Q Wouldn't that place a tremendous economic burden on the utility?

A Tremendous.

A But you would at this point elect to incur that?

A Lacking some other approach. Then after it was in operation, I would do the suggestion that you made about ten minutes ago.

Q Which one was that?

A Require every vender to -- consider requiring every vender to go back through the performance history of his design, all safety and non-safety aspects of the design, and demonstrate conformance with some sort of overall reliability criteria involving frequency of challenge to safety systems and performance of safety systems.

Q You don't think that all of that is going to place a crushing burden on the nuclear industry?

A It may.

Q You say it may. Don't you think that somebody has to give some careful consideration to that before allowing that situation to occur?

A Lessons Learned is doing that right now.

Q In terms of the economic burden to be placed on the industry?

A No--looking at what is necessary.

Q You don't see that as a function of the NRC, to assess the financial burden to be placed on the industry?

A We can't do what we require blindly. We must understand something of the practicality of what we propose.

Q Hasn't that been the view all along within the NRC that you can't do it blindly and you have got to consider burden on the industry, and as a matter of fact, I think you were explaining to me that was the consideration in regard to backfitting requirements of the SRP to plants of the vintage of TMI 2, wasn't it?

A I am suggesting what was a burden before may not be a burden today.

Q If nothing else, it is probably more of a burden since probably it costs more today than five years ago to do the same backfitting?

A The definition of what is a burden today is different than what was a burden before.

Q Okay. The burden before was substantially increased expense to the utility, wasn't it?

A - That plus some practical considerations about what the staff was capable of reviewing itself.

Q Have those practical considerations as to what the staff is capable of reviewing changed since then, since say 1975?

A Yes, because in 1975 people still thought that there would be a sizable workload from construction permit applications.

Q And there hasn't been?

A Oh,yes, there has been, since 1975.

Q In terms of the capabilities of the staff to deal with these problems in terms of lighter workload and more time to devote?

A I think they are going to have more in the future.

Q But that has not occurred since 1975; the workload has gotten heavier, hasn't it?

A Not CP workload.

Q But OL workload?

A Definitely.

Q And there hasn't been any substantial increases in the size of the NRC staff at that time, has there?

A Not substantial, not as many as are necessary.

Q Something else, well, I covered it in the deposition I have already had with Harley Silver, which was the issuance of an operating license for TMI 2 with a number of open items designated in the OL. There are some 13 to 15 items that were outstanding in connection with TMI 2, and I was curious in that Mr. Silver's comments were to the effect this is not unusual.

A That is a small number compared to what has been standard practice since then.

Q Where has the concept developed from the NRC of issuing an operating license for a facility which still has open unresolved open items relating to safety?

A Well, some of them issue an OL in order to permit fuel loading to go ahead because the issue that they affect is only of safety significance when the plant is at power. The plant is not at power when it is loading fuel. There are some that fall into that category.

Some of them in just a basic sense of equity were not found to be necessary until very late in that plant review. An unresolved safety issue was included. As an example, if they said widgets ought to be all painted blue, and later OL's are all green, and all the operating plants are green, well, here is this fellow with this billion dollars invested, as you described, and the only difference between him and an operating plant is that guy is getting a return on his dollar and this guy is not. It is going to take the same amount of time to paint them blue. If you are not going to shut other guys down when they paint, then why hold up the one fellow from operation while he is painting his from green to blue, so you give him a deadline and go into operation, and as soon as practical, he accomplishes the change. That is the concept. You don't have to argue with it.

Q I feel compelled to respond to your question why hold that guy up. If the determination is made that there is a safety factor in not having them all painted blue, why increase it?

A There is a safety factor in everything we do. There are gradations in it. There are some that are important enough to shut a plant down, and some that are not.

We like the ones even though they are not important enough to shut a plant down, so we require them to be done, we give people a reasonable time to implement them. If you give an operating plant a reasonable time to implement them and you have got another plant that is finished construction and is ready to operate, why shouldn't he have a reasonable time if it can be shown there is no radiation consequence to workers as a result of giving him the opportunity to go into operation before the change is accomplished?

Q That demonstration is required?

A That kind of review is conducted to make sure you are not causing a problem -- that you don't foreclose options by letting them go into operation. The options are not foreclosed, and the decision usually is to let them go into operation.

Q And so as a practical matter, the practice of issuing OL's with open items relating to safety has become standard practice?

A Yes. I might caution there that I have been propounding that philosophy to the Commission on the lessons learned and have run into a fair amount of static. It surprised me. Gilinsky accused me of overturning the regulatory precedent of the past where we always required more of new OL's than we did of past OL's, and my answer to that is yes, we do, but not when they are already finished construction and they are ready to operate. Then we tend to think of them as being an operating plant unless we are going to foreclose an option by letting them go into operation. They have got the same option as the operating plants. Therefore, they are at the same risk as the operating plant. My choice would be to let them go into operation.

I understand there are people who don't feel that way. You seem to be one of them.

Q I guess my attitude on that again is a very simplistic one.

A If you can't allow them to stay in operation -- if you can allow them to stay in operation and be truthful that it is not that important a safety problem, and if it is that important a safety problem, then be truthful and shut them down until they get it fixed--all of them, not just the one that is waiting to go into operation. I am arguing for consistency.

Q I understand. On the other hand, I am not so certain that consistency is necessarily what happens, that is achieved here.

A It is when you are making a very judgmental finding on safety because it is, by God, one of the few yardsticks you have got when all you have for legislation ~~to legislate~~ is

the words
no undue risk.

Q Let me ask you about something else.

A No undue risk is an historical judgment, and it changes depending upon who is exercising it,and the best yardstick I have got in making that judgment is what people have done 20 years prior to my sitting at this desk. Take that yardstick away from me and I haven't got much else.

Q We have been talking all day about the problems with that yardstick, haven't we?

A Yes.

Q Okay. Let me ask you about something else you mentioned before.

A - Don't make it worse.

Q Okay. Let me ask you in terms of the transfer of TMI from the Division of Project Management to the Division of Operating Reactors, what happens if that transfer is not accomplished after the OL is issued for a project?

It is my understanding that the TMI 2 facility was not formally transferred from the Division of Project Management to the Division of Operating Reactors right up to the day of the accident, March 28, 1979, and I believe Harley Silver told me that that has now either been accomplished or will be accomplished very shortly, but it was not done after the OL was issued in February of 1978.

In what position in terms of regulatory supervision does that place a facility when it has got its OL,but hasn't been transferred to DOR?

A It is in a diminished state, doesn't have the full attention that DOR pays to a normal operating reactor, nor does it have the full attention that DPM pays to a normal plant under construction. The focus of DPM is the granting of licenses and the priorities and resources are set to meet those ends, so it is ~~not in~~ in limbo. That is an overstatement, but it is somewhere between full attention in DPM or full attention in DOR.

To compensate for that, DOR people do begin to take cognizance of a plant before the transfer occurs by some months usually, plus DSS maintains its on-call cognizance, that is, we solve technical problems that people direct to us from the time that they first apply for a CP until it is ~~demissed~~ decommissioned, in theory. We never lose track of a plant, but we are a step removed from those licensing organizations.

Q Perhaps this relates to what we were just talking about. If I understood Mr. Silver's explanation for why the transfer did not occur, it had to do in part at least with the nature and number of the open items that were designated in connection with the OL at the time it was issued and reluctance upon DOR to assume responsibility for following up on a lot of those open items.

There was a question about how the work should be allocated and that DOR felt a bit pressed at the time and did not want to take all that added responsibility on itself.

A We resolved that resource problem about a year ago by agreement on my part to maintain responsibility for closing the open items after the transfer had occurred. That is, the technical review resources to finish the closing of open items would come from DSS and not DOR. That was DOR's reluctance several years ago at accepting transfer, but that was something I recognized fairly early as solvable, and something that we should retain responsibility for.

Q That is curious because when Harley Silver told me this question would have come up around April of 1978, which was about the time that there was a draft memorandum prepared that was designed to--

A The resolution that I am referring to probably occurred about a year ago now, last summer, early last fall.

Q A few months after, I see.

A And the fact that TMI had not transferred at that date along with some others was the genesis of reaching that agreement.

Q To prevent that situation from occurring?

A My concern was the plant laying in this status between full responsibility in DPM and full responsibility in DOR.

Q Why did it come about that TMI 2 pursuant to policy, was that transferred to DOR?

A I don't know.

Q Okay.

A Some others that were in that backlog were transferred. I am not certain why Three Mile was not.

Q All right. Let me ask you one or two more things. I am just about finished. We have received a summary of Westinghouse responses to I&E bulletin No. 7906A prepared by Mr. Anderson, the manager of the Nuclear Safety Department at Westinghouse, and on the 16th page--that is not numbered, but it has under No. 12 hydrogen gas generation and control, and it states some limits, modes for removing hydrogen and core cooling with a non-condensible gas bubble in the upper head--it is a fairly brief presentation.

I wonder if you could look at this and first of all, have you ever seen this document before?

A I am sure a copy was given to me. I haven't read it.

Q Okay. You see the reference up at the top to 50 CFR?

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

A Ten CFR, 50.46.

Q Is that a reference to some very conservative hydrogen computations that have been done in the past?

A No. That is regulation that governs the designs of emergency core cooling systems.

Q In connection with hydrogen generation?

A It says that there shall be less than 1 percent clad metal water reaction for all loss of coolant accidents.

Q All right.

A That is a limit on the performance of the emergency core cooling system.

Q I don't find any reference in those pages to recombiners being available. I wonder does that mean Westinghouse is intending to proceed without recombiners available in response to the hydrogen problem?

A Well, I can't believe that is true because Westinghouse is the only design that provides recombiners inside of containment, redundant, what have you.

Westinghouse does a good job with recombiners. They provided a recombiner design before they were required by AEC.

Q All right. I was just curious about that. Let me ask you finally about two documents you previously supplied here. One is a memorandum dated June 29, 1978 from Lawrence Phillips, Acting Branch Chief, Analysis

Branch, Division of Systems Safety, to Mr. Richard Denise, Acting Assistant Director for Reactor Safety, Division of Systems Safety.

The subject is documentation of TMI 2 benchmark calculations.

Could you just explain for the record what this document is?

A Well, there have been calculations done under the cognizance of the Analysis Branch in DSS, some scoping calculations within the branch, and other calculations by computer codes that INEL did, and Mr. Phillips and his memorandum are documenting for the record who performed what calculations, and what they showed, and comparing them with the calculations done by B&W using several of the B&W codes.

He points out where there are inconsistencies and consistencies, and points out the uncertainties in the benchmark calculations.

Q Okay. I just wanted to get an explanation on the record. I think this related to what we were talking about before.

A You had asked earlier for what kind of analysis had been done.

Q The other thing you have produced here today is a memorandum dated July 7, 1979 from Warren Minners to you, the subject being trip report, TMI systems thermal hydraulic analysis meeting, July 11 and 12, 1979, Palo Alto, California.

What is this memorandum and how does it relate to what we have been talking about?

A Again, ~~this is~~ you asked for a summary of the kind of analyses that we knew were going on for ~~at~~ the TMI 2 accident, and this is a report of a trip that Mr. Minners took to a meeting with EPRI in July where results were presented by the folks from NRC who were doing these calculations for the special inquiry, and people from INEL, and Los Alamos, presented the work that they were doing and GPU and EPRI talked about some analyses that they were doing, and it gives a summary, overview of the status of those computations.

MR. KANE: Let's have this memorandum dated June 29, 1979 marked as Exhibit 2 to this deposition, and the one dated July 17, 1979 marked as Exhibit 3.

(Mattson Exhibits Nos. 2 & 3 were marked for identification.)

MR. KANE: I have no more questions, Mr. Mattson. Mr. Chopko, do you have any questions?

MR. CHOPKO: No questions.

MR. KANE: Let me say, Mr. Mattson, I have appreciated your being available to us all day long, as you have been today. This is an on-going investigation and although I have asked all my questions at the present time, there may be facts that we will discover in the future which will necessitate having to bring you in for another deposition.

I certainly can assure you we will make every effort to avoid having to do that. However, should it prove necessary, we will be in touch with you again. For that reason, we will adjourn the deposition rather than terminate it with the thought that we may have to resume it at some time in the future, and once again, I thank you very much for your time.

THE WITNESS: You're welcome.

(Whereupon, at 4:42 p.m., the deposition of Mr. Mattson was recessed.)

REPORTER'S CERTIFICATE

DOCKET NUMBER:

CASE TITLE: DEPOSITION OF ROGER J. MATTSON

HEARING DATE: August 6, 1979

LOCATION: Bethesda, Maryland

I hereby certify that the proceedings and evidence herein are contained fully and accurately in the notes taken by me at the hearing in the above case before the

PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND

and that this is a true and correct transcript of the same.

Date: August 8, 1979

Official Reporter
Acme Reporting Company, Inc.
1411 K Street, N.W. Suite 600
Washington, D.C. 20005

PROFESSIONAL QUALIFICATIONS
Roger J. Mattson

SUMMARY

Professional experience ranges from mechanical engineering design and thermal-hydraulic analysis of research reactors and nuclear power reactors to senior line management positions in regulatory standards and power plant licensing. Activities have included routine interactions with officials of NRC, other Federal agencies, International Atomic Energy Agency, Congress, State governments, electric utility industry, commercial nuclear industry, professional societies, academic institutions, and consumer and public interest groups. Academic training through a PhD in mechanical engineering ranging from structural design and mechanical dynamics to advanced fluid mechanics and heat transfer.

EXPERIENCE

July 1977 to Present - Director, Division of Systems Safety, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Career executive position with senior line management responsibility for conduct of technical reviews and evaluations of applications for construction permits and operating licenses for nuclear power plants. Responsible for the development of regulatory requirements and policies for reactor safety, including membership on the NRC's Regulatory Requirements Review Committee and Technical Activities Steering Committee. Technical scope of the Division includes reactor systems, core performance, reactor analysis, containment systems, materials engineering, mechanical engineering, structural engineering, instrumentation and control systems, power systems, and auxiliary systems.

June 1975 to July 1977 - Director, Division of Siting, Health and Safeguards Standards, Office of Standards Development, U.S. Nuclear Regulatory Commission. Career executive position with senior line management responsibility spanning the nonengineering or software areas of the NRC standards development activities. Responsible for planning and direction of the development of regulations and other standards to protect employees of NRC licensees, the public, and the environment from the effects of activities regulated under the Atomic Energy Act, in conjunction with the National Environmental Policy Act. Technical subjects included in scope of responsibility were: energy facility siting, land use and regional planning, environmental modeling, effects of chemical and thermal effluents, emergency preparedness, economics of nuclear and other fuel cycles, geology, seismology, hydrology, meteorology, radiobiology, radiation health physics, physical security systems for sabotage and theft protection, and control and accounting of special nuclear material.

August 1974 to June 1975 - Assistant Director for Site and Health Standards, Office of Standards Development, U.S. Atomic Energy Commission until January 1975, then U.S. Nuclear Regulatory Commission. Mid-management position with scope including nuclear facility siting and environmental effects, radiation protection of workers and the public, transportation safety, and consumer product safety.

February 1974 to August 1974 - Technical Assistant to Commissioner William O. Doub, U.S. Atomic Energy Commission. Advised the Commissioner in connection with the formulation of AEC programs and policies. Acquired broad knowledge of nuclear and other energy producing fuel cycles, and of nuclear weapons, including detailed understanding of AEC policy, organization, procedures, and personnel with emphasis on the regulatory process, reactor safety, and safeguarding of nuclear materials against theft or sabotage.

February 1972 to February 1974 - Nuclear Engineer and then Section Leader, Reactor Systems Branch, Directorate of Licensing, AEC. General function was thermal and hydraulics reviews, analyses and evaluations of nuclear power reactors and associated safety systems. Responsibilities included administration of research programs. Experience led to in-depth, internationally recognized expertise in safety systems for light water power reactors, especially emergency core cooling systems.

September 1969 to January 1972 - Graduate Student and Research Assistant, Department of Mechanical Engineering, University of Michigan; on leave from AEC. Thesis in boiling heat transfer and multiphase flow for pressurized water reactors.

June 1967 to August 1969 - Reactor Engineer, Reactor Technology Branch, Licensing, AEC. Responsibilities included technical review and evaluation of nuclear power reactor safety systems and management of research programs, including emergency core cooling, reactor containment, auxiliary cooling, reactor core thermal-hydraulic design and shutdown cooling.

March 1964 to May 1967 - Technical staff member, Sandia Corporation, Albuquerque, New Mexico. Responsibilities included design of components and irradiation effects experiments for advanced research reactors and other facilities simulating nuclear weapons environment.

EDUCATION

PhD, Mechanical Engineering, University of Michigan, 1972
MS, Mechanical Engineering, University of New Mexico, 1966
BS with Distinction, Mechanical Engineering, University of Nebraska, 1964

HONORS

Meritorious Service Award, USNRC, October 1976 for management leadership in area of intergovernmental responsibilities for energy facility siting.
Special Commendation from AEC Chairman for testimony in "nuclear moritorium" suit (Nader v. Ray), July 1973
National Science Foundation Graduate Fellowship, 1971
Pi Tau Sigma - Mechanical Engineering Honors Society
Sigma Tau - Engineering Honors Society
Pi Mu Epsilon - Mathematics Honors Society
Sigma Xi - Scientific Honors Society

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

JUN 13 1979

MEMORANDUM FOR: Richard P. Denise, Acting Assistant Director for Reactor Safety, Division of Systems Safety

FROM: Laurence Phillips, Acting Branch Chief, Analysis Branch, Division of Systems Safety

SUBJECT: DOCUMENTATION OF TMI-2 BENCHMARK CALCULATIONS

Since the first few days after the TMI-2 accident, several calculations were performed in the Reactor Analysis Section of AB. The purpose of this memo is to document these calculations and to point out conclusions and questions which may be considered for further study by the task forces now assigned to the TMI-2 review.

(1) Core Uncovery Calculations

Based on studies of the plant process data which were plotted for the period immediately after the accident, it was concluded that core uncovery first occurred at 105 minutes after turbine trip. This was indicated by primary coolant temperature data and steam generator pressure data which indicated a loss of primary to secondary heat transfer, therefore inferring cessation of primary coolant circulation and condensation in the steam generator. Subcooling of primary coolant had ceased at approximately sixty minutes after the event, and all reactor coolant circulation pumps were turned off by 100 minutes.

Calculations were performed to estimate the necessary primary coolant mass discharge rate to achieve top of the core uncovery at 105 minutes. The calculations were based on the RCS pressure history and the assumptions which follow:

(1) No HPI coolant after approximately eight minutes.
(2) Mass discharge rate varied as the square root of the pressure times density product,
(3) All liquid mass above the core exit elevation (including pressurizer) was discharged at 105 minutes.

Results of these calculations are indicated in Figure 1 and show that a mass discharge rate of 395,000 lbm/hour (rated @2250 psi) would be required. This compares to a rated steam discharge rate of 118,000 lbm/hour @2250 psi through the power operated relief valve (PORV).

Since the required mass discharge could not be achieved by steam flow through the PORV alone, the effect of varying the discharge quality using various critical flow models was studied. These results are indicated in Figure 2.

It was concluded that very nearly 100 percent liquid discharge and little or no HPI flow for the 100 minutes preceding core uncovery would be required to explain the system mass loss by discharge through the PORV alone. Since the new HPI flow (in excess of letdown flow) is believed to have been 30 to 100 gpm or more during this period and since continuous discharge of very low quality coolant through the PORV for this entire interval is highly unlikely, a leakage source other than normal PORV discharge is inferred by these calculations. Additionally, later compilations of sequence of events data show that the Reactor Building Sump high level alarm (4.650 feet above the bottom of the sump) was received at about eleven minutes after turbine trip. Since the rupture diaphragm on the reactor coolant drain tank did not burst until fifteen minutes, PORV discharge does not explain the high building water level.

The core uncovery calculation was continued based on steam generation as a function of the decay heat rate and the portion of the core covered. The predicted core steaming rate during the core uncovery interval is given in Figure 3 based on (a) mass and energy conservation calculations and (b) mass and volume conservation calculations. The corresponding plots of core water level are given in Figure 4; an early B&W estimate of the core water level is also indicated on the plot. The plots indicate that the minimum core water level during this interval was no more than three feet and may have been below the bottom of the core.

Corresponding core heat-up calculations during the uncovery interval were performed using TOODEE. These calculations indicate that the clad melting point occurred in advance of total zirconium oxidation. More details of the core heat-up results will be documented separately.

(2) Once Through Steam Generator Heat Sink Capacity

Calculations were performed to estimate the effects of PORV setpoint and reactor trip response on the response of once through steam generators to the Loss of Feedwater Transient, assuming that no auxiliary feedwater is available. Design data for the Midland plant were used as the basis for the calculations. Three cases are tabulated in Table I. Case 1 assumes pre-TMI setpoints for PORV pressure relief and for reactor trip. Case 2 assumes a reduction in the overpressure reactor trip setpoint to 2300 psig, versus a 2450 psig higher setpoint for the PORV which may preclude its opening during the transient. Case 3 further reduces the time to reactor trip by providing a reactor trip on turbine trip.

An energy balance was performed for the first 110 seconds of the transient, which was the time required to boil the steam generator dry for Case 1. Steam generator heat transfer response and the minimum primary coolant temperature of 552F corresponding to secondary steam pressure control conditions were taken from B&W calculations of the transient. Decay heat rates are based on 1971 ANS data with a best estimate multiplier of 0.9. A feedwater coastdown of 10 seconds is assumed.

For Case 1, the primary heat sources including stored energy released when the primary system drops from 582F average temperature to 552F provide sufficient energy to boil dry the 109,000 pounds of mass in the steam generator and to produce 592F superheated steam at an assumed saturation pressure of 1020 psia. However, it is important to note that the steam generator dry out process has reduced the energy level of the system so that 538 seconds of decay heat would be required to return the primary system to initial average temperature of 582F. Therefore, initiation of auxiliary feedwater could be delayed up to nine minutes without rise in the primary system energy level above the initial level.

Case 2 results in about one second earlier trip time and corresponding reduction in the full power seconds generated after turbine trip. Three percent of the steam generator coolant inventory remains available at 110 seconds, and approximately 585 seconds of decay heat are required to restore the system to its initial energy level at 582F.

Case 3 results in instantaneous trip with less than one second at full power. The primary coolant system is reduced to the assumed minimum temperature of 552F (secondary saturation temperature is 547F) with 29 percent of the secondary coolant inventory remaining. Approximately 17 minutes of decay heat are required to boil the balance of steam generator coolant and restore the system to its initial energy level of 582F.

It can be concluded that the early trip time is equivalent to increasing the steam generator coolant inventory by: .29 x 109,000 = 31,610 pounds. Typical steam generator coolant inventory and boil-off data were computed for loss of ac/dc Power Task Action Plan A-30 as follows:

Plant	Type	Power (Mw)	S.G. Mass (lb.)	Time to Boil Dry*
Midland	B&W	2552	92000	17 minutes
St. Lucie	CE	2570	253000	61 minutes
Zion	W	3238	357146	70 minutes

*Assumed 1971 ANS decay heat with no multiplier.

It is clear that an early reactor trip setting is of little significance to the post-accident response of CE and W steam generators with large coolant inventory, even if auxiliary feedwater is not available. For B&W steam generators with low water inventory (probably less than the design value), an early trip would significantly delay the occurrence of steam generator dry out and provide more time for initiation of auxiliary feedwater.

It is also noteworthy that the steam generator coolant inventory depletion can be delayed by higher secondary pressure relief setpoints (CE vs. B&W). However, higher secondary pressure also limits the primary cooling at a higher temperature level, so that less time is required to reheat the primary system if the heat sink is lost. Therefore, there appears to be no ultimate advantage to higher secondary pressure relief setpoints.

(3) Evaluation of TMI-2 Benchmark Calculations

It was noted that both B&W and INEL benchmark calculations produced overcooling of the primary system during the first two minutes of the transient. An energy balance based on plant process data was performed by hand calculations for the intervals (0-110) sec., (110-300) sec., (300-360) sec., and (360-540) sec., in order to better understand the indicated transient and reasons for the error in computer models.

Table II is a tabulation of the bases and results for the Reactor Coolant System (RCS) energy balance calculations.

The steam generator mass, including feedwater added during a linear ten second coastdown, was depleted during the (0-110) second interval. Mass inventory of 55,970 pounds was computed from the measured level of 160 inches based on a shell side flow area of 44.4 ft.^{2} in each steam generator. This compares to a B&W reported mass of 97,000 pounds which is believed to have been used in their benchmark calculations. The difference of 41,030 pounds is believed due to the difference between actual heat transfer performance and design heat transfer performance. Better performance results in lower liquid level in a once through steam generator. The additional mass would correspond to an added heat sink of approximately 30,000 Mw-sec and would lead to an over prediction of the Reactor Coolant System cooldown during steam generator dry out. In fact, the LOFW analysis would look like Case 1 of Table I with the RCS cooled to 552F compared to the measured temperature of 577F. The sensitivity of safety analyses to assumed mass inventory of the steam generator should be considered in future review of plants having once through steam generators.

The energy balance during the first 110 seconds resulted in excess energy of 1495 Mw-sec. Decay heat energy is believed to be at least as great as the estimate and possibly 10 percent more. Uncertainty in the number of full power seconds or in the steam generator heat sink could account for the deficit. For the purposes of the tabulation, the amount of steam that could be generated by the excess energy of 13.6 Mwt each second is assumed to flash and discharge through an unidentified leak (e.g., a steam generator tube).

One peculiar aspect of the TMI-2 data is the mismatch in reactor coolant cold leg temperatures at the start of the transient. Loop A was at 563F while Loop B was at 557F. The control system would normally be expected to maintain a much closer match between these temperatures; e.g., ± 2F, and the reason for this condition should be investigated.

The (110-300) second time interval should be ideal for an energy balance. During this interval, the HPI was known to be operating at full capacity and was the only heat sink other than normal heat loss to the containment. The RCS average temperature was nearly constant during this interval. The only heat sources were decay heat (estimated uncertainty of -0 + 10%) and the reactor coolant pumps. This heat balance shows a large deficit of 5219 Mw-sec or 27.5 Mwt per second. The only plausible explanation of the deficit is flashing of the primary coolant. Since at least 10F subcooling is indicated during this interval, flashing could only occur at a leak location. Leakage through the stuck open PORV would be supplied by flashing in the pressurizer, which was analyzed separately from the RCS heat balance. Leakage flow rate for this energy deficit would be 47.3 pounds per second of steam, compared to 25.9 pounds per second for the deficit indicated during the first 110 seconds.

The balance was continued for the interval of (300-360) seconds when the RCS temperature rises to 562F and reaches saturation temperature. An energy deficit of 12 Mwt per second (equivalent steam leakage of 19.3 pounds per second) is indicated for this interval. However, there was greater uncertainty in the HPI coolant injection rate and in the energy supplied to the reactor coolant system during the reheating. A calculation performed for the six to nine minute interval with RCS at saturation and rising in temperature showed only a 3.6 Mwt/sec deficit, indicative of very little flashing during that period. The latter result is surprising and possibly indicative of lower quality leakage from the saturated system. However, the calculations may be in error after six minutes since the pressurizer cannot be properly separated from the RCS heat balance after saturation is reached.

Table III is an energy balance of the pressurizer to evaluate the calculated leakage and the calculated level based on the system pressure history during the first six minutes of the transient. Time intervals were chosen to match Table II except that no balance was made after six minutes when saturation temperatures was reached. An equilibrium pressurizer model was assumed with flashing energy supplied by all of the hot fluid in the pressurizer at the beginning of a calculation interval. RCS water was not included in the balance.

Steam relieving rates through the PORV were normalized to a discharge pressure of 2250 psi for comparison of discharge capacity during the three time intervals and for comparison to the rated relief capacity of 118,125 lbm/hr. Excellent agreement with rated capacity was obtained, particularly during the (110-300) sec. period when the calculational uncertainty is at a minimum. This tends to confirm that pressurizer leakage was via the stuck open PORV and an additional RCS leak is needed to explain the mass balance.

Also of interest is the computed liquid level versus the measured liquid level. The calculations indicate that the indicated level was too high by 35 inches at 110 seconds, 151 inches at 300 seconds, and 137 inches at 360 seconds. These calculations are believed to be reasonably accurate and indicative of substantial error in the liquid level reading during the first several minutes of the transient. The high indicated levels suggest flashing in the reference leg during the depressurization.

In summary, the following conclusions were reached from the benchmark calculations.

(a) The B&W computer model for CADDS benchmarking of TMI appears to have several deficiencies; e.g., too much water inventory in the steam generator; 3% heat demand to simulate HPI cooling effect from two to five minutes (this is too much); etc., which could be pursued to obtain a more acceptable benchmark.

(b) The INEL model had several problems with improper handling of auxiliary feedwater being the largest error contributor. They are now aware of these problems and are making appropriate corrections.

(c) All calculations seem to point to leakage in addition to that through the stuck open PORV.

(d) A reactor trip on turbine trip has the same effect as additional inventory in the steam generator and appears to be of no value for plants having steam generators which reduce the primary temperature to near the secondary saturation temperature without boiling dry.

(e) Safety analyses sensitive to steam generator coolant inventory should be reviewed carefully to assure that conservative water levels are used.

(f) There remain sufficient questions about the TMI-2 response to warrant additional benchmarking analyses.

Laurence E. Phillips

Laurence E. Phillips, Acting Branch Chief
Analysis Branch
Division of Systems Safety

Enclosures:
Tables

cc: F. Schroeder
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TABLE I

PRIMARY COOLANT ENERGY RELEASE TO BOIL DRY

THE STEAM GENERATORS

Reference: Midland Data	CASE 1 (Pre-TMI)	CASE 2* (Post-TMI)	CASE 3** (Post-TMI)
PORV Set Pressure (psig)	2340	2450	2450
Time to Reactor Trip (sec)	8.93	8.0	0
Time to Rod Movement (sec)	9.63	8.7	0.9
Feedwater Flow Rate (lbm/sec)	3500	3500	3500
Feedwater Added During Coastdown (lbm)	17000	17000	17000
Steam Generator Secondary Water (lbm)	92000	92000	92000
Total Coolant Mass Boiled (lbm @ 1035psig)	109000	109000	109000
Avg. Reactor Coolant System Temp. @ 0 sec. (°F)	582	582	582
Avg. Reactor Coolant System Temp. @ 110 sec. (°F)	552	552***	552***
Avg. Secondary Steam Pressure (psia)	1020	1020	1020
Estimated PORV Flow (avg. lbm/sec)	-	-	-
Power Level (Mwt)	2552	2552	2552
PRIMARY HEAT SINKS (Mw-sec)			
(a) Steam Generator	74157	74157	74157
(b) PORV	NEGLECTED		
PRIMARY HEAT SOURCES (Mw-sec) to 110 sec. after Turbine Trip			
(a) Full Power (2552 Mwt) Energy Input	24575	22202	2297
(b) Fuel Stored Energy (1350F→550F)	12913	12913	12913
(c) System Stored Energy (582F→552F)	25392	25392	25392
(d) Decay Heat to 110 sec.	9297	9297	10057
(e) Reactor Coolant Pumps (18 Mwt) to 110 sec.	1980	1980	1980
TOTAL ENERGY SUPPLY TO S.G. @ 110 sec.	74157	71784	52539
FRACTION OF STEAM GENERATOR COOLANT NOT BOILED	0	.03	.29
TOTAL DECAY HEAT FOR ADIABATIC HEATING OF THE SYSTEM TO 582F ENERGY LEVEL	346[illegible]9	37062	56967
TIME (sec.) required to generate decay heat	538	585	1006

*Overpressure trip @ 2300 psig; S.G., would not be dry @ 110 sec.

**Reactor trip on turbine trip; S.G. would have substantial inventory @ 110 sec.

***Primary coolant temperature cannot be lowered below 552F due to limiting secondary saturation temperature of 547F which limits further heat transfer.

POOR ORIGINAL

TABLE II

TMI ENERGY BALANCE BENCHMARK

Basis: Plant Data Describing the Accident

	(0-110) sec	(110-300) sec	(300-360) sec	(360-540) sec
ORV Set Pressure (psig)	2,255	OPEN	OPEN	OPEN
ime to Reactor Trip (sec)	9	--	--	--
ime to Rod Insertion (sec)	10	--	--	--
eedwater Flow Rate (lbm/sec)	3,180	NONE	NONE	NONE
eedwater Temperature (°F)	463	--	--	--
eedwater Added During Coast-down (lbm)	15,897	--	--	--
team Generator Coolant Inventory	55,970	NONE	NONE	NONE
otal Coolant Available (lbm)	71,870	NONE	NONE	NONE
team Superheat Temperature (°F)	592	--	--	--
vg. Secondary Steam Pressure (psia)	1,035	950	865	825
vg. Reactor Coolant Temp. @ 0 sec (°F)	582	582	582	582
inal Reactor Coolant Temp. Avg. (°F)	577	578	582	595
vg. Primary Steam Press. (psia)	1,900	1,560	1,375	1,425
Estimated Leak Flow (Avg. lbm/sec)	26.9	47.8	19.8	6.1
PI Coolant Added (Gallons)	243	2,136	200	600
Level (Mwt)	2,688	--	--	--

RIMARY HEAT SOURCE (Mw-sec)

	(0-110) sec	(110-300) sec	(300-360) sec	(360-540) sec
a) Full Power (2688 Mwt) Energy Input	26,880	0	0	0
b) Fuel Stored Energy (1360°F - T Coolant)	12,590	0	-81	-212
c) System Stored Energy	4,250	-850	-2,800	-9,100
d) Decay Heat	9,792	13,290	3,604	9,970
e) Reactor Coolant Pumps (18 Mwt)	1,980	3,420	1,080	3,240
OTAL ENERGY SUPPLY	55,492	15,860	1,804	3,398
raction of S.G. Coolant not boiled	0	--		
OTAL DECAY HEAT FOR ADIABATIC EATING OF THE SYSTEM TO 582°F NERGY LEVEL	14,042	--	--	--
ime (sec) Required to Generate Decay Heat	175 sec	--	--	--

RIMARY HEAT SINKS (Mw-sec)

	(0-110) sec	(110-300) sec	(300-360) sec	(360-540) sec
a) team Generator	52,610	0	0	0
b) HPI Coolant Heating	1,167	10,261	961	2,882
c) System Heat Losses	220	380	120	360
d) Deficit	1,495	5,219*	723	656
ENERGY Deficit per sec (Mwt)	13.59	27.47	12.05	3.64

Fc 100% quality leakage from the RCS (excluding the pressurizer) to compensate the energy defic
*Note that a lower average energy discharge indicates that lower quality coolant is being discharge.

POOR ORIGINAL

TABLE III

TMI-II PRESSURE VOLUME/ENERGY BALANCE

		(0-110) sec	(110-300) sec	(300-360) sec
PZR Pressure (psig)		2150 - 1745	1745 - 1400	1400 - 1355
Coolant Properties:	V_g(ft^3/lbm)	.1685 - .2275	.2275 - .2974	.2974 - .3094
	V_f(ft^3/lbm)	.0265 - .0245	.0245 - .0232	.0232 - .0230
	h_g(BTU/lbm)	1123.3 - 1153.3	1153.3 - 1172.6	1172.6 - 1175.
	h_f(BTU/lbm)	689.1 - 641.1	641.1 - 600.7	600.7 - 594.7
Electric Heater Input (Mw-Sec)		180.18	311.22	98.28
Pressurizer Volume per ft. Level Change (ft^3/ft)		38.48	38.48	38.48
Reactor Coolant Average Temperature (°F)		582 - 577	577 - 578	578 - 582
Fraction of Hot PZR Fluid Flashed		.0937	.0706	.0103
LIQUID VOLUME (V_f), Ft^3		795.35 - 604.4	604.4 - 811.2	811.2 - 932.4
DELTA LIQUID VOLUME (V_f)				
RCS Shrinkage, Ft^3		-99.4	0	+99.4
PZR Shrinkage, Ft^3		-60.0	-32.1	-7.0
Liquid Flashed, Ft^3		-68.9	-40.4	-5.3
Liquid Boiled by Heaters Ft^3		-8.6	-12.5	-3.7
Liquid Added to RCS by HPI Ft^3		+45.9	+291.9	+37.8
STEAM VOLUME (V_g) Ft^3		704.65 - 895.6	395.6 - 688.8	688.8 - 567.6
STEAM MASS (W_g)				
Steam at Beginning of Interval lbm		4131.9	3936.7	2316.2
Added by Flashing, lbm		2812.6	1741.7	231.4
Added by Heating, lbm		350.	544.1	161.3
Steam at end of Interval, lbm		3936.7	2316.2	1834.5
Balance lost thru PORV, lbm		3407.8	3906.6	874.9
Steam Relieving Rate (lbm/Sec)		31.0	20.6	14.6

TABLE III (Cont'd)

Equivalent Relief Rate (lbm/Hour) @ 2250 psi		134150	114502	92279
PZR LIQUID LEVEL:	Measured (in)	220-195	195-376	376-400
	Calculated (in)	220-160.4	160.4-224.9	224.9-262.7

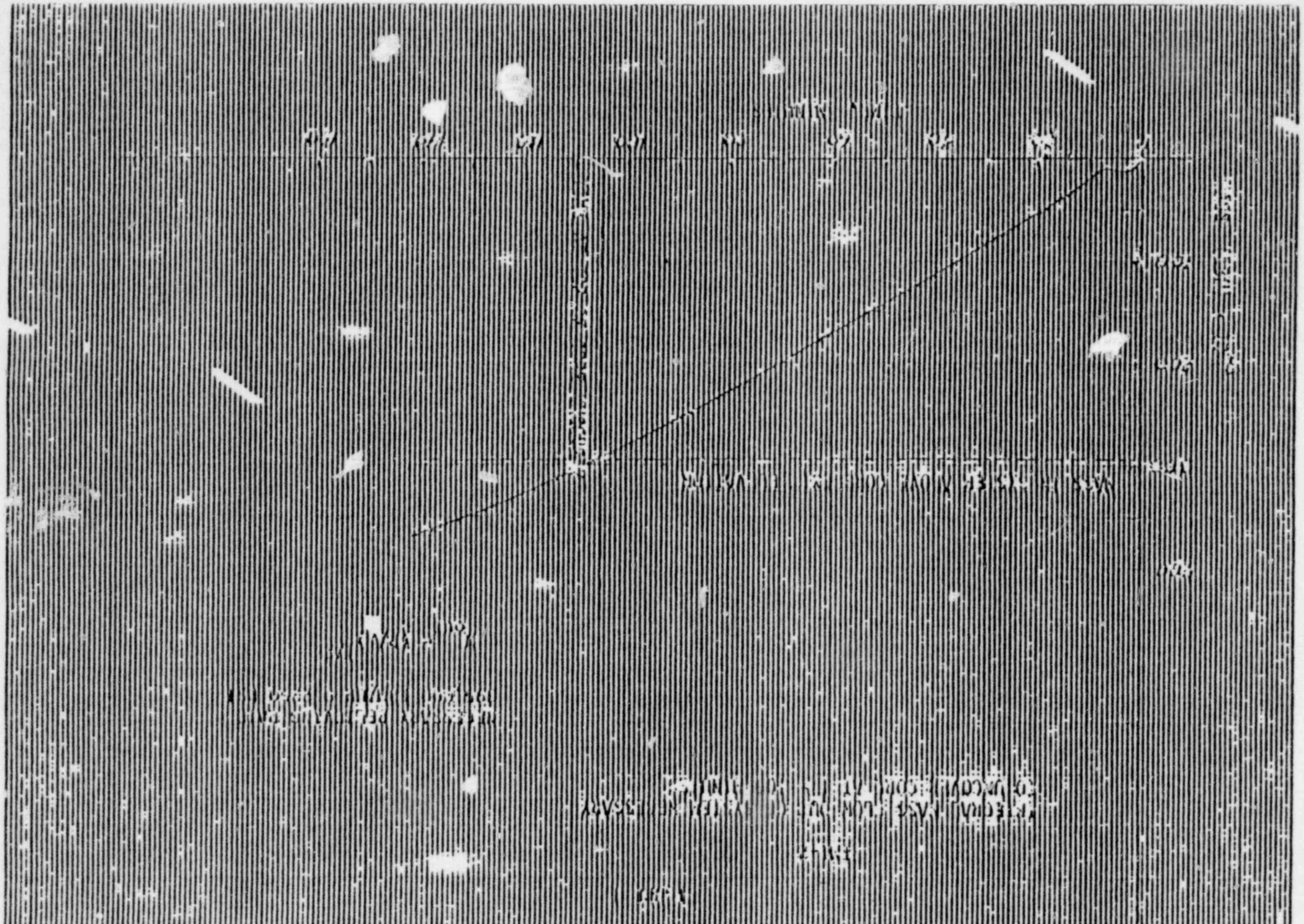
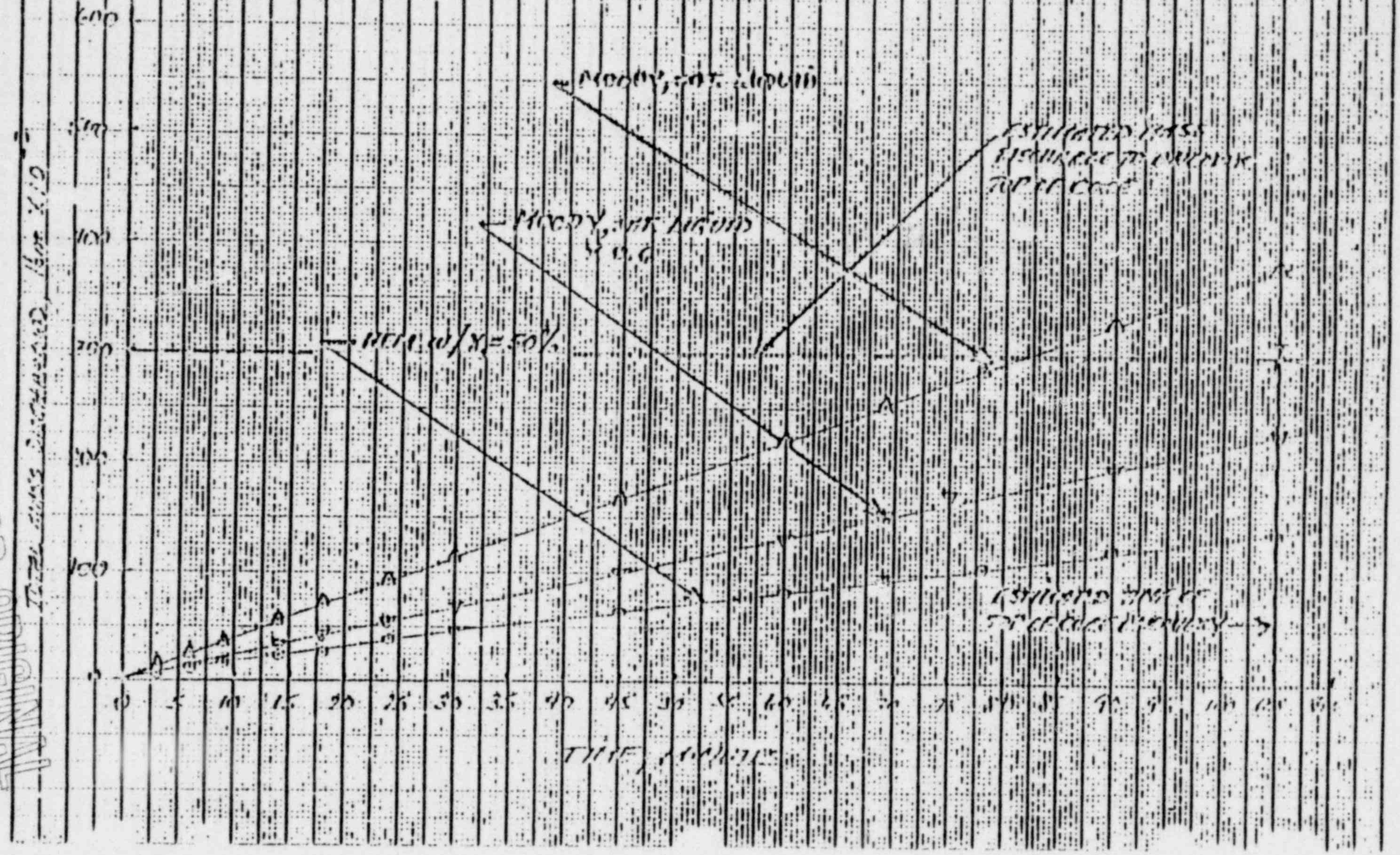
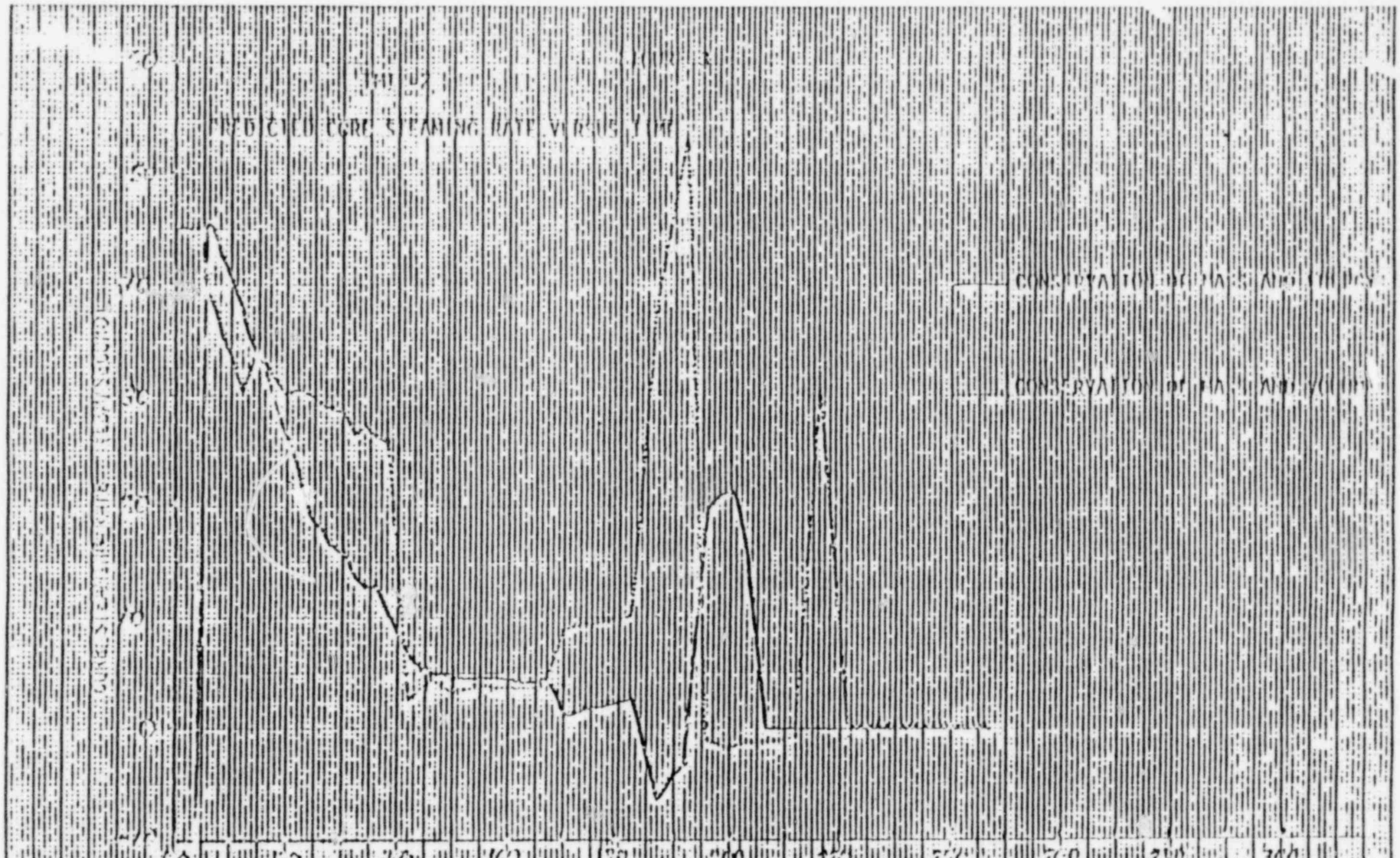


FIGURE 3 TMI-2

MASS DISCHARGED FROM SYSTEM AS FUNCTION OF TIME FOR VARIOUS CRITICAL FLOW MODELS AND SATURATION CONDITIONS




PREDICTED CORE STEAMING RATE VERSUS TIME
TIME, MINUTES

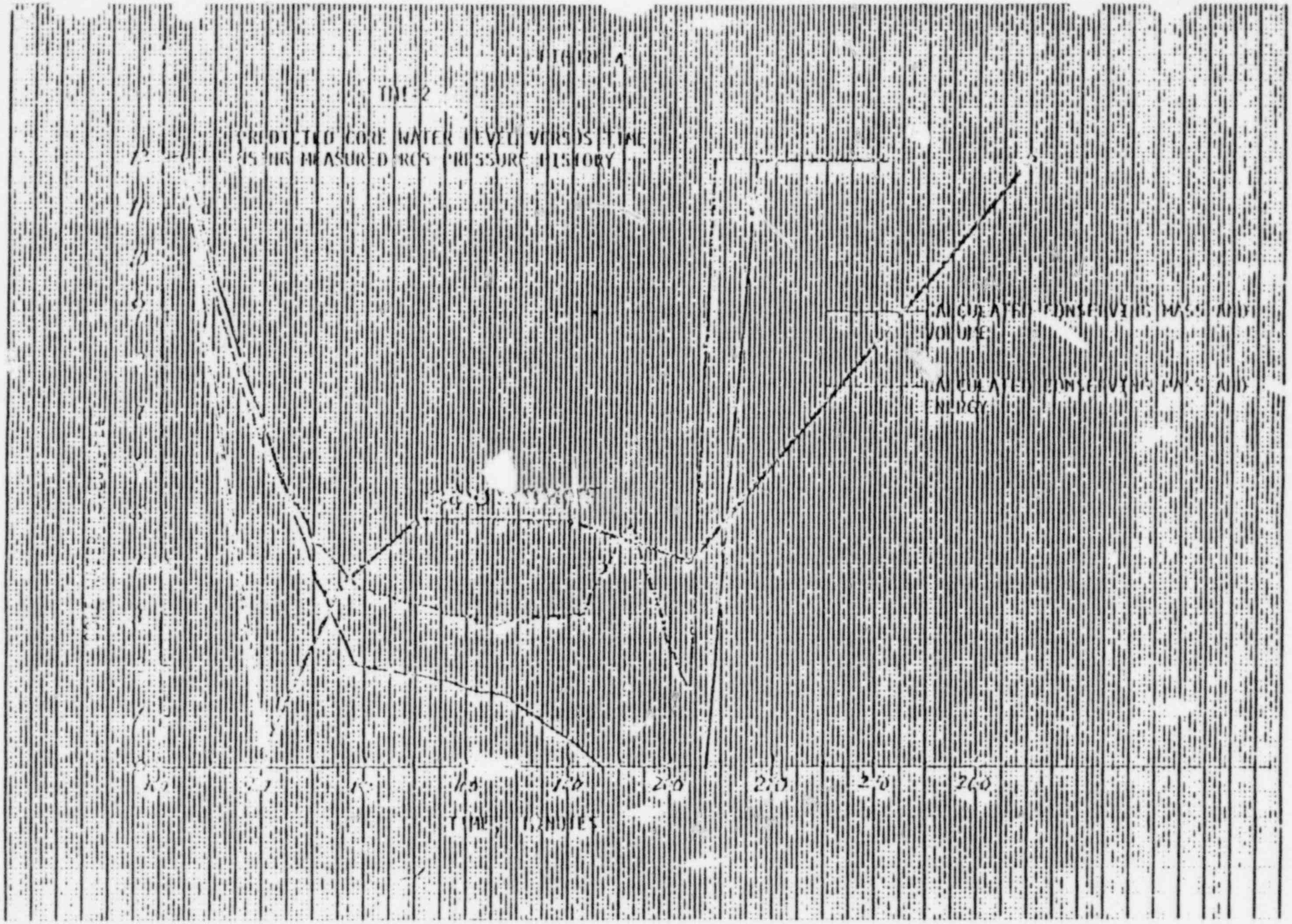


FIGURE 4
TMI-2
PREDICTED CORE WATER LEVEL VERSUS TIME
USING MEASURED RCS PRESSURE HISTORY
CALCULATED CONSERVATIVE MASS AND VOLUME
CALCULATED CONSERVATIVE MASS AND ENERGY
TIME, MINUTES

JUL 1 7 1979

MEMORANDUM FOR: Roger J. Mattson, Director
TMI-2 Lessons Learned Task Force

FROM: Warren Minners

SUBJECT: TRIP REPORT - TMI SYSTEM THERMAL HYDRAULIC ANALYSIS MEETING JULY 11 & 12, 1979, PALO ALTO, CA

The meeting by EPRI with the objectives of 1) gaining additional technical insights on the TMI-2 system thermal-hydraulic response, 2) to assess the capabilities of current computer programs and 3) to identify the improvements in the computer programs. The meeting was attended by representatives of all of the organization (EPRI, GPU, ITI, INEL, LASL and NRC) analyzing the TMI accident data. An agenda listing of the participants is attached. Copies of the slides presented at the meeting are available in my file.

Two general approaches for evaluating the TMI-2 system response, data analysis and computer analysis, were presented.

Data Analysis

Both EPRI and GPU presented their analysis of the data. The majority of the discussion centered on the first 100 minutes of the accident, that is until shortly after the reactor coolant pumps were shut off.

EPRI concludes based on an energy balance on the relief valve drain tank that the relief valve failed in the full open position. However, the mass flow rate through the valve is sensitive to the steam quality and has not yet been determined. Based on their computer analysis, B&W calculates twice the mass flow rate estimated by EPRI. Both B&W and EPRI estimate the flow rate in the letdown line to be about 140 gpm based on a heat balance across the letdown line cooler. Therefore the letdown line flow is comparable to the flow from the relief valve.

However, B&W estimates that the net letdown/makeup flow averaged a positive 35 gpm which sets the makeup flow at 175 gpm. EPRI believes such a high makeup flow should be revealed by the response of system parameters which they do not see. EPRI estimates the makeup flow to have been 40 gpm, which result in a negative net letdown/makeup flow of 100 gpm. There is a consensus that the water level in the RCS after the RCP were turned off was not above the pressure vessel hot leg nozzles.

EPRI speculates that since the system appears to have begun to repressurize before the relief valve block was first closed, hydrogen may have been generated soon after the RCP were stopped. EPRI also speculated that a rapid 100°F increase in the cold leg temperature and similar 300 psi increase in system pressure at 3 hours and 45 minutes may indicate the time when the core collapsed. However, these increases can be also explained by increases in HPI flow, outflow from the pressurizer and re-wetting of the core. After generation of hydrogen little or not heat transfer through the steam generators occurred and cooling was due to flow of HPI injected water through the core. During the period between 11 and about 16 hours there was no significant heat removal from the core.

Comp[illegible] Analysis

Computer analysis of the accident are being made to 1) evaluate the capability of the codes, 2) fill in gaps in knowledge of the accident and 3) investigate the effect of possible alternatives. The NRC representatives urged that the primary emphasis be placed on the last purpose, that is answer some questions of "what if".

The codes being used are RETRAN (GPU, ITI and EPRI), RELAP (INEL), TRAC (LASL) and CRAFT (B&W). The RETRAN codes used a detailed system model (about 100 volume(including modeling much of the secondary system. The RELAP and TRAC models used half as many volumes (about 50) and much simpler secondary side models. Except for TRAC which had 3-D modeling of the vessel, the models were one dimensional.

The computer analyses generally predicted the trends of the parameters better than the quantitative valves. The codes that attempted to model the secondary system in the most detail (RETRAN) made poorer predictions, than codes that used secondary conditions as boundary conditions to a reactor coolant system (CRAFT).

The time span calculated to date is short as shown by the following table.

CODE	INVESTIGATOR	TIME SPAN
RETRAN	GPU	8 min.
RETRAN	ITI	3 min.
RETRAN	EPRI	1/2 min.
RELAP	INEL	100 min.
TRAC	LASL	100 min.
CRAFT	B&W	100 min.

The calculations show that the relief valve mass flow is sensitive to the steam quality. All analyses over[illegible] the initial [illegible] in pressurizer level. The RCS flow rate decay due to void formation is under-predicted by the two codes (RELAP and TRAC) for which the parameter was available.

POOR ORIGINAL

In order to facilitate completion of the analyses and comparison of the results a "standard" set of initial and boundary conditions was developed based on the best information available to date.

Original signed by
[illegible]

[illegible]

cc: LLTF Dist.
PDR
Z. Rosztoczy
D. Ross
W. Johnston
S. Fabic

ENCLOSURE:
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