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NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF:

THREE MILE ISLAND SPECIAL
INQUIRY DEPOSITION

DEPOSITION OF:

DR. DENWOOD F. ROSS, JR.

POOR ORIGINAL

Place - BETHESDA, MD.

Date - Friday, September 28, 1979 Pages 1 - 109

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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In the Matter of: :
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THREE MILE ISLAND :
SPECIAL INTERVIEWS :
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DEPOSITION OF DR. DENWOOD F. ROSS, JR.

Room 6110
Maryland National Bank Building
Bethesda, Maryland

Friday, September 28, 1979
9:05 a.m.

APPEARANCES:

For the Nuclear Regulatory Commission:

WILLIAM PARLER, NRC/TMI Special Inquiry Group
C. O. MILLER, NRC/TMI Special Inquiry Group
DENNIS ALLISON, NRC/TMI Special Inquiry Group
TTHOMAS COX, NRC/TMI Special Inquiry Group

** ** **

C O N T E N T S

EXHIBITS

** ** **

P R O C E E D I N G S

Whereupon,

DR. DENWOOD F. ROSS, JR.

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

EXAMINATION

BY MR. PARLER:

Q Please state your full name for the record.

A Denwood F. Ross, Jr.

Q Dr. Ross, did you receive a letter --

A Yes, I have a copy with me.

Q -- providing you with important information concerning your deposition?

A Yes, I did.

Q The copy that I show you I will assume is a photocopy of the letter which you have received; is that correct?

A Yes, that is correct.

Q Dr. Ross, I have marked that letter for identification as Exhibit 1151.

A Okay.

MR. PARLER: The letter that is so marked is a letter dated September 6, 1979 to Dr. Denwood Ross from Mitchell Rogovin, Direcotr, NRC/TMI Special Inquiry Group.

(The document above referred to was marked for identification as

Exhibit 1151.)

BY MR. PARLER:

Q Dr. Ross, do you understand the information that is set forth in this letter, including the general nature of the NRC/TMI special inquiry, your right to have an attorney here today as your representative, and the fact that the information you provide here may eventually become public?

A Yes, I understand that.

Q Dr. Ross, is counsel representing you personally today?

A Is?

Q Yes, is.

A No.

MR. PARLER: I would like to note for the record that the witness is not represented by counsel today.

BY MR. PARLER

Q Dr. Ross, if at any time during the course of this interview you feel that you would like to be represented by counsel and have counsel present, please advise me and we will adjourn these proceedings immediately to afford you the opportunity to make the necessary arrangements.

Is this procedure agreeable to you, sir?

A Yes, it is.

Q Dr. Ross, you should also be aware that the testimony that you give has the same force and effect as if you were testifying in a court of law. My questions and your responses are being taken down, and they will later be transcribed. You will be given the opportunity to look at that transcript and make changes that you deem necessary. However, to the extent that your subsequent changes are significant, those changes may be viewed as affecting your credibility. So please be as complete and accurate as you can in responding to my questions. If, at any point during the deposition, you don't understand a question, please feel free to stop and indicate that, and we will make the necessary clarification at that time before we proceed.

Dr. Ross, did you bring a copy of your resume to this deposition?

A Yes, I did. I would like to note that this is slightly abridged from the resume that I gave the Presidential Commission, because the prior resume didn't include the job assignment I had from October '78 to the present. So this has been updated.

Q That is fine, Dr. Ross.

A This is the resume.

MR. PARLER: I will mark for identification as Exhibit 1152 a one-sheet resume headed at the top "PERSONAL QUALIFICATIONS, DENWOOD F. ROSS, JR."

(The document above referred to was marked for identification as

Exhibit No. 1152.)

THE WITNESS: I would like to make one addition to this. I would like to enter today's date, 9/28/79.

MR. PARLER: Off the record.

(Discussion off the record.)

MR. PARLER: Back on the record.

THE WITNESS: I made the abridgements to my prior resume. This was typed today. So the date of 9/28/79 should appear somewhere on this sheet of paper.

MR. PARLER: I have marked in the upper righthand corner "9/28/79." Is that all right?

THE WITNESS: That is correct.

BY MR. PARLER:

Q Now, at the present time, Dr. Ross, your assignment is what, sir?

A I have been detailed to be the Director of a task force within NRR known as the Bulletins and Orders Task Force.

Q And prior to that your assignment was what, sir?

A Prior to that -- this assignment was effective in June of this year -- I was the Deputy Director of the Division of Project Management in the NRR.

Q I realize that your resume speaks for itself, but I have one final question. Prior to your being assigned the position of Deputy Director, Division of Project Management, what was your position, and for approximately how long?

A I was the Assistant Director for Reactor Safety in the Division of System Safety, NRR.

Q And in that position, sir, what generally was your responsibility and what groups did you supervise?

A There were are are three branches in the collection known as Reactor Safety. The branches are the Core Performance Branch, the Reactors Systems Branch, and the Analysis Branch. The general function of those three branches, which I collectively supervised, was to study the reactor engineering aspects of reactor safety. This includes the physics, fuel performance, the thermohydraulic performance, and transient and accident behavior.

Q Dr. Ross, you were scheduled for a deposition today primarily for the purpose of providing information on the assignment that I have been given on the Special Inquiry Group. That assignment is to look at the regulatory process for the licensing and regulation of nuclear power reactors as it functioned prior to March 28, 1979, that is, prior to the TMI-2 accident.

We have with us at this deposition today Mr. C. O. Miller, who is a consultant to the Special Inquiry Group, who also has a different area of interest and responsibility as a consultant to the group. He will now proceed to ask questions to you, Dr. Ross.

After Mr. Miller finishes, Mr. Dennis Allison also has some questions which he will ask you in still a different area.

After these two gentlemen finish, I will then proceed with you, sir, in the areas relating to the regulatory program that I am interested in and have the responsibility for.

Mr. Miller, will you please proceed, sir.

MR. MILLER: Off the record.

(Discussion off the record.)

MR. MILLER: Back on the record.

BY MR. MILLER:

Q Dr. Ross, as Mr. Parler mentioned, my area of inquiry here is a little different from his, and indeed Mr. Allison's, and I would summarize it as being safety engineering and management aspects of NRC operations.

In this context, please don't hesitate to ask me to clarify any of the questions I ask, the terminology, and , indeed, if it is more convenient for you to cite a reference which will simplify your testimony and lead me to information in a more straightforward manner than your testimony, please don't hesitate to do that also.

Finally, I have read your Presidential Commission testimony and similarly, if there is something in there that you wish to call my attention to, just mention it and I will do the necessary research when the time comes. I don't believe we will have any occasion to refer to it specifically unless you want to.

A Okay.

Q Let's begin, then, with a topical matter based on a newspaper article yesterday, but I had the question in my own mind before that.

I wonder if you could differentiate for us between licensing and regulation as applied to the NRC's process if, indeed, there is a difference.

A I believe there is a difference. I read the same article you did.

I would think that licensing would be to develop or expand on present standards and apply them to plants not yet built, so it would have the combination of new and different standards and new and different plants, whereas regulation would be to apply present standards to plants already built.

That is a rough definition, but it is as good as any.

Q All right. Are there any methods of achieving nuclear power plant safety that NRC employs, other than what might be described as being under the regulatory or licensing process?

A Well, the Commission, the NRC, has a very large research budget, in excess of $200 million a year. The Department of Energy also spends some money on safety, but I don't know how much.

I imagine that that has an influence on design. It's a little hard to attribute directly how this might affect design and safety, but I am sure it has an effect.

I will give one or two examples.

This agency, NRC, has funded research at several places around the country to more precisely define the rate at which heat is continued to be generated after the reactor shutdown, sometimes called TKE, sometimes called after-heat production. Presently, in the regulatory sense, there are some conservatisms applied which affect the way plants are run. It appears that the research that the NRC sponsored will permit relaxation of some of the conservatisms and permit more favorable plant operation, perhaps higher power, perhaps for a longer period of time.

That is one example. I think that the money spent on research would be an indirect effect. And that is not through any application of regulatory regulations, design criteria, regulatory guides, or standard review plans, which is the traditional way that plants are both licensed and regulated.

Q Do you know of any programs, current or past, in which the NRC effectively became an educational-type force to achieve nuclear power plant safety, such as conducting seminars, training people outside NRC, or things of that sort?

A Well, I have been involved with an international program where one of the missions that I had was to export reactor safety information. This agency is a participant in the Committee for the Safety of Nuclear Installations. The acronym is CSNI. The CSNI is a subsidiary of the OECD, which is an international group based in Paris, which does a lot of things, one of which is reactor safety. There is a working group part of the CSNI known as the Ad Hoc Working Group on Emergency Core Cooling, and I was an official in that working group. Anywhere from about 12 to 20 countries participate. And we have had seminars here that I would arrange and bring in consultants from around the country to instruct the other countries as to how our analysis tools work for analyzing loss-of-coolant accidents.

Q Would it be a reasonable conclusion for me to draw that you are probably typical, in a sense, of many staff members of NRC who participate in professional committee activities, or groups of that nature?

A Not quite. The activity I was just describing was what I would call an active thing, where I was actively preparing a lecture series for the international group.

Q I see.

A I think a scientific committee or industry committee, that I have never worked on, is more of a group effort, where each person brings in his share and contributes. There are a large number of NRC people on standards committees, industry-based committees, but I have never been on one and I am not sure exactly how they work.

Q I see. Thank you.

A I should point out one other thing the NRC does routinely. Each year this agency has, in the latter part of

October or early part of November, a week-long conference at the National Bureau of Standards. It is called a Water Reactor Safety Research Conference. Several hundred people come from ll over the world while the NRC Research Office presents seminars and lecture series on the research activities of the last year.

Q Speaking now of the various NRC decisions that emanate from design or operational reviews, does the NRC policy in these matters acknowledge the utilities' desire to -- to use their words -- save the shutdown whenever possible during transients?

A I guess I really don't understand that question.

Q Let me expand on it somewhat. Some of the documents we have reviewed in our inquiry group reflect -- or you can assume for the purpose of my question that they reflect -- the utilities' desire, for economic reasons, not to shut down the reactor if they can avoid it during certain transients. And my question is to what extent, perhaps, is this phenomenon accounted for in the regulatory process by NRC.

A I understand.

The desire to avoid unnecessary shutdown is reflected in writing in, among other places, Chapter 15 of the Safety Analysis Report. An example might be that if the reactor is running, say, at full power and there is a sudden decrease in the desire of the electrical grid to absorb the electricity that has been generated, what the control system wants to do is to tell the turbine not to work so hard and not to generate so much electricity. In turn, the turbine tells the reactor to quit producing so much power.

So the control system accommodates all of these things. It slowly reduces reactor power steam water rate, so as to accommodate what the turbine is generating to what the grid wants to absorb. This whole thing is sometimes called a runback.

If you can run back the reactor quickly enough with the control system, then you won't have to shut the reactor down prematurely.

This would be described in various transients that are part of Chapter 15, such as sudden load rejection, and the reason would be clearly stated, yes. The burden of the licensee would be to show that the runback can be absorbed without having to scram in order to meet our regulations.

Q Thank you, Dr. Ross. May I mention on the record that I, for one, truly appreciate your simple explanation of this, because I hav had my difficulties with this in the past.

A Yes.

Q You know, I am sure, that the statutes governing NRC --

A What was that word?

Q I'll rephrase the question. I'm sure you will appreciate that the statutes which gave rise to NRC, particularly the Energy Act, established a Commission, and established certain key offices, NRR, NMSS, and, I believe, RES was the other one.

A That's right.

Q I wonder if you could, within that context, offer any views on how safety decisions are made, that is, safety decisions by the offices vis-a-vis safety decisions reserved for the commissioners as a body or, for that matter, individually.

A Well, most of the significant safety decisions, I believe, are made at or below the assistant director level. The design of the reactor, the containment, and the safety systems, are described either in what comes into the building, which is called a safety analysis report, or described in topical reports. That is the incoming material.

The material that leaves the building is the staff safety evaluation report, called an SER. So the incoming is an SAR, which the reactor or the reactor vendor provides, and the outgoing is the SER.

The SER within the Division of Systems Safety is generally prepared by a staff member, reviewed by his section leader, if he has one, his branch chief, and it is transmitted under the signature of the assistant director to the Division of Project Management, who then collates all of the inputs and publishes it.

The decision, for example, to accept a new emergency core cooling system feature or a new containment feature, a new structural design feature, would be made at or below the assistant director's level. So by and large I would say that that is the area where the prominent decisions are made.

Now, some of the more fundamental, let's say, deciding-policy decisions, for example, would you want to put a reactor in downtown New York, would certainly not be made at that level. That decision would be made at the Commission level, I'm sure.

Decisions to reject a site would be made, I'm sure, at the office director level.

But the great majority of the safety decisions are made at the assistant director level.

Q Expanding on this somewhat, in the sense of going not only between the three offices I mentioned before, but also including the Office of Inspection Enforcement and the Office of Standards Development, to what extent are these decisions made autonomously within a given one of these five major offices?

A Of course I will just have to qualify this answer to my understanding.

The only way that I see safety decisions being made collegiately by the five offices is through what is colloquially known as the Ratchet Committee, more correctly known as the Regulatory Requirements Review Committee.

This committee has delegates from those five offices that you mentioned. Its chairman is Edison Case of NRR. In the area of standards development, who would have a new standard or regulatory guide, or from the office of NRR, who would want to modify the way that we presently license plants through the standard review plan, if we wanted to increase the requirements on the industry, which is the colloquial ratchet, then there is a formal process for bringing that matter to the attention of the Ratchet Committee. A lot of the important decisions are made there. I shouldn't have used the word "decisions." The Ratchet Committee proposes and the Office, Director of NRR disposes. That is, he is the final arbiter of whether a new requirement is a policy matter with respect to licensing.

That is the only decision process involving those offices that I am familiar with.

Q Could you expand a little bit -- you touched on it a moment ago -- on the role of the Division of Project Management, first in relationship to other divisions within NRR, and secondly -- we can come back to this in a minute -- in respect to the other major offices besides NRR.

So first, would you care to comment, please, on the role of the Division of Project Management in the decision-making process now, in relationship to other divisions within NRR.

A The Division of Project Management has a decision-making role in that as it receives questions to be sent to the applicant, and during the conduct of a safety review, it is supposed to insure consistency that these question areas or positions that are proposed to be taken are consistent with present policy.

As an example, if one of the reviewing branches requested Project Management to notify a utility that henceforth he must have six pumps instead of two, the project manager should be familiar with present policy, and if that constituted a ratchet he should advise the offending branch to proceed to the Ratchet Committee, because that is our policy.

Another decision area he faces when he is assembling the safety evaluation report is to look for the same sort of thing, to make sure that the document the agency is going to publish is consistent with present policy.

Now, there is an additional decision-making thing. As each applicant comes in with a request for a construction permit or an operating license, he is informed that during the course of the review if he disagrees with the matters that are decided upon, he has the right to make an appeal. He may appeal a requirement to a two-man committee, which is composed of the Division of Project Management and the Director of the Division of Systems Safety. So the Director would have a decision-making role.

There are activities inside the Division of Project Management that are not directly related to licensing of construction permits or operating licenses. There is a branch known as the Operator Licensing Branch which serves all reactors, not just those under review. Occasionally it is necessary for the

Operator Licensing Branch to deny a license to a person to run a reactor. That denial would be subject to review by the Division Director, an additional decision-making process. And in that context the Director would have a decision-making process on new requirements for licensing.

I think that about covers that.

Q Let me expand just a little bit. What if there became a difference of view between, say, the Director of the Division of Systems Safety and the Director of the Division of Project Management? What mechanism arises for this to be resolved?

A I can only speculate. They would consult the Office Director.

Q And logically, as an organization segment, if it came to that, he would make the decision?

A That is correct.

Q All right. Let's go on, then, a little bit into, as you see it, the role of the Division of Project Management as compared to offices outside of NRR. Are there any potential areas of conflict that would have to be resolved between these groups on occasion?

A I don't think so. From time to time the Project Management people might ask the Office of Standards Development for an interpretation of a general design criterion, which is Appendix A to the Commission's Part 50 regulations. I don't know of any interaction with NMSS, and very little interaction with the Office of Research. There might be some interaction between Inspection and Enforcement and the Quality Assurance Branch, because Inspection and Enforcement has a heavy role in construction quality assurance, and so does the Quality Assurance Branch. I don't have the details, but controversies could arise there, because both parties are trying to achieve the same thing.

Q What is the dividing line between the Division of Project Management and the Division of Operating Reactors?

A If you start at time zero when a utility decides it wants to build a reactor and files for a construction permit,the Division of Project Management is the interaction between the technical branches and the utility. During the construction phase, after it has received its construction permit, it continues to be the official point of contact.

The plant then files for an operating license, and again Project Management is the point of contact. You receive this license, and for the first few months of operation Project Management is responsible for, say, servicing the license, in other words, providing license amendments, to the point when the reactor seems to be functioning well. It is then transferred to the Division of Operating Reactors.

The point of contact, the responsibility for servicing the license, is transferred.

At that point there is an interfacing memorandum issued that is jointly acceptable to both division directors, that officially and formally hands off the project.

So a new project manager in DOR would be assigned, a new branch chief, and Division of Project Management would have no further responsibility.

Q As a practical and, indeed, physical point, when this transfer takes place, what happens to the documentation associated with the work under Division of Project Management? Does it essentially be handed over, or files be pointed out to DOR and they say, "Here it is," or just what takes place?

A Of course, most of that I can't answer. I don't know whether the project manager physically hands the thing over. The memorandum of transfer is frequently quite lengthy. It may have 20 or 30 items undone, and each item will be definitive with respect to who is going to do what, and on what time scale. Because there is never a clean transfer; there are always some IOUs, work to be done in the future.

But as far as the files are concerned, I don't know if the project manager hands over his diary, his log book, his personal files, or not.

Q Let me give you an example, and perhaps that will help clarify matters.

In the design and development of a nuclear power reactor, unless I am mistaken, the contractor or the utility or someone might perform some extensive hazards modes and effects analysis or failure modes and effects analysis. These are conceivably reflected in some report or other, but they might not be. In other words, there will be correspondence and documents that reflect what went on. And what I am trying to get a feel for is: How is this information, if you will, translated between the DPM and DOR?

A Well, if it is something that is done, it would have been reflected in the staff safety evaluation report. If it is not done but needs to be done, it will be in the memorandum of transfer.

Q I see. If I use the term "matrix concept of management," would you understand what I mean, or would you like me to expand it?

A I think I understand it.

Q I am curious to know as to whether NRC's fundamental approach to management, reflected in DPM or DOR, utilizes a matrix concept of management to staff and carry out their programs.

A Not as I understand the term, no.

Q Would you explain to me your understanding of the term "matrix management."

A If we used matrix management, what it would mean to me is we would have collections of people, for which we can use the term "branch" -- would have branches of people with like disciplines. And each branch would have a manager, and maybe collections of branches would have an assistant director, such that you had an assembly line of technical disciplines.

The project manager then, for each project, would be assigned one or more members from each discipline, and for the duration of the project he would provide technical direction. And they would not receive technical direction except maybe in the most rudimentary se. e from their technical branch chief.

The matrix would mean -- for example, the columns would be the branches, and the rows would be the project managers.

Q And I believe your answer to my question was that NRC does not necessarily function that way; is that right?

A They don't do that because the project manager has little or no authority to manage.

Q Is this why in your Presidential Commission deposition you stated that the technical decisions were made by the other branches or offices, whatever they were, other than Program Management?

A That is correct.

Q This area that we have been talking about, project management -- was this any different under the Atomic Energy Commission, to your knowledge?

A It was markedly different. In the time span from 1967 through 1971, where the regulatory aspects of civilian nuclear power were with the AEC -- the NRC did not exist -- I was a project manager, and as a project manager I had considerably more authority and autonomy than a project manager does today.

The reason is not that the management system changed so much as that the types of people, the numbers of people available today greatly exceed those available then.

At that time many of the systems branches that we have now simply didn't exist. There was no Reactor Safety Organization, there was no Reactor Systems Branch, there was not an Analysis Branch. And many other branches just simply didn't exist. So the project manager had to pick up these functions more or less on his own.

Q Are you saying, then, they operated more on a matrix concept then than they do now?

A No, it is not that they had the people; it is just that the project manager himself performed these functions. So the project manager might have to do his own review of emergency core cooling system, whereas now it is done for him; he might decide more pumps and more valves are needed, whereas now it is done for him. It is not that he had more people to manage then, because there were less people; it's just that he had more work to do.

Q Are you aware of any studies or considerations given under NRC to go more to the matrix concept of management as you defined it?

A I know that there was a report which is known as the Pocock Report. Pocock was a consultant to Mr. Rusche when he was the Office Director of NRR. I think I read the report. I know I discussed it. This would have given more stature to the project manager and given him more authority. I would have to re-read it to be sure it went into the matrix -- I believe it did go more into matrix management.

Q As a personal opinion, do you feel that more of a matrix concept would be a better way to do things at NRC?

A No, sir.

Q You are satisfied with what we have?

A Well, I am not satisfied with what we have. I think the project manager has been relegated somewhat to a paper shuffler. I think he needs to not only take more of a technical interest in the project, but to be more of a part of the decision-making. But I think that matrix management divides the responsibility too much. I would give him more technical voice, but I wouldn't want him to circumscribe the limits of the review.

Q The project manager, as he is known today -- is he required to have, if you will, both technical and managerial skills to qualify for that position?

A I think by and large yes. Project managers, when they are hired in, may not necessarily have demonstrated any managerial skills. However, in order to continue to grow -- the effective project manager acquires the skills and is promoted; the ineffective one doesn't acquire them and isn't promoted.

Q To make sure the record is clear, when you say "skills" you mean managerial?

A Managerial.

Q And technical?

A I think the skills that are looked for are more on the managerial side: the ability to weld together a cohesive report and get the work out on time, get the right people in the right place at the right time. This takes requirements more than being a detailed technical expert.

Q When, as well as how, does a project manager get involved with a new facility?

A At present he may get involved up to a year before the ulitility makes a filing. It is one of the lessons that was documented in a report NUREG-0202, such that a year in advance of the filing of the application, the project manager may be working with the utility, making sure the utility understands what should be in his application, having meetings at the site so the local people will know a nuclear plant may be coming and they will have a chance to say something about it.

Q . that early stage is it safe to assume a project manager may be essentially assigned more than one facility?

A He would be, yes.

Q Do you have any personal familiarity with the

American Society of Safety Engineers?

A I have never heard of them.

Q How about the National Safety Management Society?

A Never heard of them.

Q Next, the Systems Safety Society.

A Never heard of them.

Q And lastly, the Human Factor Society.

A Never heard of them.

Q I show you a couple of documents here. I don't think we need to enter them into the record because I am not going to question you on them other than to ask you if you have, first, ever been exposed to a document entitled "MILSTANDARD 882-A," dated 28 June 1977. And its full title is "Military Standard System Safety Program Requirements."

A I don't recall ever seeing this. I worked at the General Dynamics Fort Worth facility from 1957 to 1967. I can't preclude something like this trickling down through the Air Force because General Dynamics was an Air Force contractor, but I don't recall.

Q Does the number MIL S-3130 mean anything to you?

A No.

Q Did you ever know a Mr. William Funk with General Dynamics?

A No, never heard the name.

Q A second document I would identify similarly as

"NASA Safety Manual, NHB 1700.1(V3), Volume 3, System Safety."

A No, I have never seen it.

Q Have you ever been exposed to a program known as the MORT Program? And I show you, by way of illustration, a document entitled "MORT, the Management Oversight and Risk Tree," put out by the AEC with a code of SAN 821-2.

A No, never seen it before.

Q Within NRC, which organizational segments, to your knowledge, are concerned with man's impact on power plant safety? And I will expand that slightly to say "man" meaning control room operator, maintenance personnel, or those who are associated with the operation of the system.

A Okay. There are several groups.

One of the requirements for getting a construction permit or an operating license is to demonstrate that there are going to be enough people working at the station to do everything that has to be done. This is described in the application, and there are NRC requirements. I forget which chapter of the application it is -- it is either Chapter 12 or Chapter 13. It would describe the station manager, the technical services people, the maintenance people, and so on, the total number of people that would be running the plant. And there are NRC requirements, both in numbers and educational qualifications.

The control room operator, which we call the licensed operator, must meet standards that are described in part 55 of the Commission's regulations, and the standards are administered by the Operator Licensing Branch of the Division of Project Management. So these people physically go to the plants, go to the simulators, give oral and written examinations to prospective operators.

There is a third area where the Inspection and Enforcement Office reviews at the facility the procedures by which the plant is operated and determines that the operating staff understands and implements the procedures correctly.

I think that is the three areas.

You mentioned control room. We don't look at the human-factor aspect of the control room: Is the control room layout a decent layout? Can a guy run it? That is not formally done anywhere.

Q Do any formal or informal safety boards or councils exist within NRC, by whatever name, and if so, what levels of organization are reflected in their membership?

A The only such board would be the Ratchet Committee that I previously mentioned, whose primary function is to review new requirements.

Q Is there any group -- again, I stress informally or formally -- that would be expected to meet to review the impact of a serious event, accident, or whatever?

A No. There is no standing board.

Q Are program reviews that DPM conducts ever devoted exclusively to safety issues?

A I don't guess I understand. The work that is done by Project Management is to determine whether a construction permit or operating license should issue. And what we are supposed to find out is consistent with the Atomic Energy Act, which is: Is the plant safely run? The whole thing is safety, unless you are talking about environmental.

Q Let me rephrase the question and see if I can point out what I am getting at here.

Certainly any program, any project, in having review meetings will have many subjects on the agenda, everything from how does the schedule look to making a decision of clearcut design safety importance. What I am trying to establish is whether or not as a matter of practice within NRC there is any other form of a program review meeting which is more clearly delineated as being a safety program review meeting, as distinct from what I have just described at the beginning of my question here as a normal program review meeting.

A The only thing I can think of is that somewhere in the environmental area the need for power is discussed. Maybe that's it. But I am not familiar with that.

Q All right.

Considering the organization at NRC to achieve its objectives towards nuclear safety, what path does the line management function follow? And I will define "line management function" as being the safety decision-making function.

I know you answered earlier that certain decisions were made at a given level. Perhaps if you would just carry that on all the way up as high as necessary -- and to simplify my question, what path does the line or safety decision-making function follow throughout the entire NRC?

A We have already discussed how the review is brought together and collated at the assistant director level, who would collate his several branches.

Project Management would assemble the whole thing into a report, in other words, a safety evaluation report, which would be concurred in up through -- it would be transmitted to the Advisory Committee on Reactor Safeguards at the DPM assistant director level.

At this point we have to go outside the office of NRC and go to the next phase, which is statutorily required, go to the Advisory Committee for Reactor Safety, the ACRS. They review the report, if it is a construction permit, and write a report to the Commission. So they have a voice. They provide advice to the Commission.

The matter then would be -- if there is a hearing involved -- taken up by an Atomic Safety and Licensing Board, and we would do it through the public hearing phase. At such time as the hearing board reported, the Project Management assistant director would sign the authorization to proceed. The Office

Director is not involved, the Commission is not involved, the Executive Director for Operations is not involved in the decision-making process.

I am couching all of these things in the time period before March 28.

Q Yes; my question certainly relates to that.

A All right.

Q But let me go a little further into a couple things you said there. I believe you specifically said that ACRS provides advice. Is the same thing true of the Licensing Board you mentioned?

A That would authorize the agency to issue a permit.

Q So that is a decision-making function.

A But there is a de facto decision-making by the advisory committee, because if they fail to advise, then I don't believe we would proceed. If they advised to the negative, then that would have to be ironed out somehow. So there is a de facto decision-making process by the advisory committee.

Q If the project manager, following the process you just mentioned, made a decision and this came to the attention of the head of NRR, could he overrule it?

A Oh, certainly.

Q Could the Executive Director for Operations overrule it?

A I don't know.

Q Could the commissioners collectively overrule it?

A Oh, I am sure that they could, because collectively they are the supervisor of NRR.

Q Could the Chairmar individually overrule it?

A That I don't know. Mr. Denton, in his testimony earlier this year, noted that he felt that the Commission collectively were his boss. I'm sorry; I just have to rely on what he said.

Q Certainly. Let me interject at this point that if your response needs to be qualified, don't hesitate for a minute to do so.

I would like to ask a related question.

MR. PARLER: Off the record.

(Discussion off the record.)

MR. PARLER: We'll take a short recess.

(Whereupon, a short recess was taken.)

MR. PARLER: All right, let's continue.

BY MR. MILLER:

Q Dr. Ross, to pursue a question similar to the one I just asked you, other than ACRS, which you have already discussed, are there any other persons or organizational segments within NRC that provide a staff safety function -- and I will define "staff safety function" as being an advisory one.

A I can't think of any, no.

Q In your judgment, is there any one person or any one office who is most responsible for decisions related to safety?

A The office -- let me split that in two parts.

I think the two offices that are jointly most responsible are Inspection and Enforcement and NRR. Inspection and Enforcement, by virtue of having a large complement of people in the field, are more readily familiar with the day-to-day problems, and they have a day-to-day decision as to whether operations should be terminated.

NRR also has a responsibility for operating reactors, as well as those planned or under construction.

And I am not sure I could separate it. I would say the two of them together probably have about the same amount of responsibility.

Q As a program manager, when you look across the entire spectrum of NRC's organization, do you have any office in mind that you turn to for advice relative to safety? I am trying to speak now in terms of most responsible for advice related to safety.

A I will have to give two answers.

If it is a routine project and we are trying to decide whether the project should be licensed or not, and, if it should be licensed, under what terms and conditions, then I think NRR is the prime office. And I don't think, generally speaking, that we would go outside of the office for advice.

If it were an event of a serious nature, such as Three Mile

Island, and we got an immense amount of advice, the Research Office and the army of consultants that they have at their disposal.

Q From your perspective, do you see any particular organizational segment in NRC more active in the accident prevention business than any other particular organizational segment?

A I guess I'd have to turn back to NRR and say that that is more nearly their function than any other office.

Q All right. To what extent does the organization of NRC parallel the organizations that you see at utilities or their major contractors? Or, to put it another way, could a person, particularly in your area, sort of look to these other activities and find an opposite number, so to speak?

A If we look at the major contractors, they are known as the nuclear steam supply system contractors.

Q If I may interject, B&W being a typical example?

A Yes. Then you would see a logical complement. They have project organizations and technical organizations, and there is a very close one-to-one correspondence.

The utilities -- not so much for the utilities, because the utilities have a wide spectrum of talent. Some utilities are small, do not maintain the large engineering organizations, and rely extensively on external advice from other firms like architect-engineers.

Other utilities staff up and maintain a large internal engineering organization. Even so, even a large utility with a large organization does not have one-to-one correspondence, because much of the work, even then, is delegated back to the nuclear steam supplier, particularly in the reactor safety areas like physics, fuels, and thermal hydraulics.

Q How would you characterize Met Ed in the context of the answer you just gave?

A My answer will be in terms of two or three or four years ago, which is when I was more actively involved with that organization.

My recollection is they didn't have a strong engineering organization. It would have been more the former than the latter.

An example of a utility that did have a large engineering organization would be Duke Power, or TVA.

Q What variables do you feel are most frequently encountered that enter into a decision in matters relative to safety?

A What was the second word?

Q What variables.

A Since I didn't get the second word, read the question again.

Q Let me give it to you again:

What variables are most frequently encountered when you are faced with a decision in matters relative to safety?

A Two general headings would be probability and consequences, which collectively make the risk.

Is that too general an answer?

Q No.

A Okay.

Q I would like to add one other question here, though, and it might suggest that you amplify your previous answer.

I would be interested to know if you believe implementation of safety changes or, for that matter, tasks to achieve safety, always cost somebody money.

A I would have to say almost always. There may be some changes -- I can recall some changes that we have made in safety areas that resulted in additional power-generating capacity which benefited somebody.

Q Would it be safe to say that depending on whether you are looking at short-term goals or long-term goals, the cost-effectiveness could be a little different?

A I guess I probably don't really understand the question, but it would seem to me like cost effectiveness has to include long-term performance or it is incomplete.

Q Then to what extent, if any, are safety decisions at NRC made using cost-effectiveness as one of the variables?

A The word "cost-effective" is used from time to time perhaps in a loose, perhaps in an incorrect sense. But the actual technique of determining whether an improvement in safety merits the added cost is not rigorously done.

Q What forms of communication and documentation are used to forward safety decisions, warnings, or similar indicators of action related to safety? I know you have already spoken in terms of SERs and SARs. My question here is based more on a day-to-day basis: A problem comes up, it is resolved, and some form of communication goes out to advise somebody of what action has been taken.

Can you provide any further descriptions of those kinds of communications or documentation?

A Well, it takes several forms. I didn't bring the examples with me.

If a utility reported a problem -- and their reporting might be because the license required them to report it -- then we would have coming in a letter, perhaps a telegram, describing the problem.

If the problem had generic implications, then the Office of Inspection and Enforcement might issue a circular or an information notice to tell other people of this potential problem. And there are hundreds of examples that could be shown on that.

We might need to meet with the utility to better understand it, in which case there would be a meeting notice, which has wide distribution, both within and without the agency.

If we decided to direct the utility to alter its operations, we would send the utility a letter. And once again, this letter has wide distribution.

Somewhere in that sequence of events, if it appeared appropriate, we would notify the licensing boards if this was new or different information. On many occasions we would issue a press release that this incident had happened, or new information had come out.

I guess that is the flow of paper, if that was your question.

Q Would what you call the technical content of those things you just described usually emanate from within NRR?

A Either NRR or I&E, one or the other.

Q All right.

A I&E is equally capable of writing the information notices. There is even an earlier thing known as a PN, a preliminary notification, which I&E writes within a few hours of something happening.

Q Now, perhaps you are not the right person to ask this question, but I would appreciate your views in any case, because I am sure you must have seen literally hundreds, if not thousands, of these things.

I would like to know whether these documents contain a description of the hazard which is really being protected against, as distinguished, say, from telling somebody to do thus and so.

A Frequently they do. The most likely thing would be in the staff minutes of meetings where the subject is explored in more detail. A preliminary notification usually is only a half-page long, and it is a factual report, not an analysis. If the reactor tripped and a pump didn't start, that is what it would say. It wouldn't say what the safety significance is.

The outgoing letter from us to the utility could ask for an engineering analysis. So from time to time material coming in from the utility would have it. Most likely the minutes of the meeting would be the most detailed.

Q Are you aware of any internal NRC procedures which define the content of these things in the sense I just asked, in other words, to be sure that the communication describes what the hazard is, as well as the directions accordingly?

A Yes, there are detailed procedures on notification and exactly what is supposed to be in there.

Q Where would I find those?

A The procedures?

Q Yes.

A Mr. Vassalle, Mr. Dominic Vassalle, is the keeper of all these records, and would have all that information.

Q Is there a priority system described anywhere into which, say, categorization of some of these communications would fall?

A Not really. There have been some recent changes that I am not familiar with on board notification, on timing. You have got to tell them within a certain time, and I don't know what that time is.

However, if you have a meeting and you want minutes, there is no priority that says, "Get them out in seven days," or some fixed time.

Q Certainly the output of your Ratchet Committee would have, in a sense, priority because it might impose a time limit, would it not?

A They do. More frequently now than in the past, the Ratchet Committee would give precise deadlines.

Q I guess my earlier question was aimed at trying to find out if there is any system in existence which says, "This is a Priority A change; this is a Priority B change," in some form like that.

A Well, the Ratchet Committee has three categories:

Category 1 is for new plants, but not existing plants.

Category 2 is, "We don't know." It's done on a case-by-case basis. We might do it on existing plants or we might not.

Category 3 is everybody.

There is no time in that.

Q I guess I didn't make my question clear enough. I was curious whether any safety priority was established, such as extreme hazard, hazardous, well, maybe hazardous, that sort of thing.

A I understand. There was a crude ranking done of the unresolved safety issues. There were about -- well, there were over a hundred unresolved safety issues documented -- I don't know the exact date, but it was about two years ago. There was a joint effort between NRR people and the Probablistic Assessment Staff of Research to prioritize these issues in a descending order of safety importance. And some numerical system for quantifying this ranking was constructed solely for that purpose.

The Ratchet Committee considered the prioritization. They noodled a little bit with some of the way they were ranked, and finally approved them, and a top 20 came out of that.

The agency focused on that, and some of the bottom ones were so unimportant they were just dropped altogether. The ones in the middle would be worked on a time-available basis.

I think that is more responsive.

I believe Mike Acock of NRR is most familiar with that, if you want more information. I could get it; I just don't have it with me.

Q Thank you very much. It is very helpful.

In that same vein, however, do you believe that these risk assessment or hazard probability studies that you have seen from time to time provide you a high level of confidence that the hazards that have been looked at are really understood?

A To the extent I have used them, yes. My task force has used these techniques in the last few months to make safety decisions, and they have been very helpful. They point out the dominant contributors to risk so that we can work on those and eliminate those. They have been very helpful.

Q You are sayming "predominant contributors." May I conclude from that that you are thinking of the substance as opposed to the numerical values thereof?

A We were influenced by numerical values. Numerical values were assigned for the operator to perform an action, or in an active malfeasance where he executed an action he shouldn't have, the likelihood that a valve had failed to operate, and so on. Numerical values were put in to decide what the dominant contributors of a certain system were to liability.

Q What activity was that you are just describing now?

A It was done under my present detail assignment. We were looking in a probablistic sense at the performance of the auxiliary feed water systems of all of the pressurized water reactors.

Q Is this that you have just described reflected in any documentation, a report of any sort?

A Yes. There is a draft report that is being reviewed that describes the methodology. And the application to operating reactors is reflected in a number of letters we have recently sent to operating reactors. I don't have any with me, but they are available.

Q Can I ask you to provide for the deposition subsequently identifying materials, things which describe the process you have just discussed here in your deposition?

A Yes.

Q What process is followed to record and track safety deficiencies determined from accident incident investigations, either by -- well, whoever.

First of all, what system is available to do this?

A At present there isn't a system.

Q Is there any system in existence which records and tracks recommendations that people have made resulting from accident or incident investigations?

A Not really. There may be one in what I would call a docket sense. If there is an operating reactor and it is investigated and as a result it is decided to require the licensee on that particular docket to do something so that the accident doesn't happen again, then the docket will reflect all of this. If it had generic implications, then Inspection and Enforcement bulletins or circulars or information notices might be sent to everybody that might have this disease.

But the system that you are talking about is -- I understood it and I answered it in the sense, "Do you have a follow-up system to be sure that all plants responded to a request to determine if they had the disease, and if they had it, to fix it?"

No, there is no system that does that. It may be done on special cases; it may be done by perseverence of a project manager. But a tracking system, no.

Q Let me define a difference between three kinds of visits that might be made to a utility.

I am going to define an inspection as one in which certain predetermined requirements are examined to see if the utility adhered to them.

I am going to describe a staff assistant visit as being one in which qualified people go out and basically discuss or otherwise try to help people solve a particular problem that may exceed their capability in-house.

Thirdly, I want to define a safety survey as being the use of a white-hat inspection. That is, they will go out with a team of people and evaluate what is going on, like an inspection with one major difference: There is no penalty involved in it if they find something wrong, and indeed it is a method whereby the communications are only to the very senior people of the organization, on the basis that they in turn would not use that information for disciplinary reasons.

My question is: In your knowledge of NRC, have all three of these, or what proportion of these three methods have been used as part of a safety management program?

A Well, the first one, 90-plus per cent; the last one, to my knowledge, not at all. The second one, maybe a few per cent, but not very much.

Q To what extent does a project manager become involved in the adequacy of the management functions at a utility or at a major contractor?

A Fairly high. One of the findings that the Act requires us to make is that the applicant is technically qualified to construct and operate a facility. That determination is mostly made by the project manager in consultation with other people. There is not much in the way of written guidance on how one comes to that decision. The Quality Assurance Branch helps, but by and large the project manager has to do that on his own.

Q I think you have just answered my question, but to make sure I understood you correctly, there are no written criteria for the management approach to this sort of thing as you would require of a contractor or utility?

A No, there is nothing like a standard review plan or an industry standard for determining that a utility and its contractors are technically qualified. It is largely subjective, based on experience.

However, in a recent adjudicatory project on the Sharon Harris application, we developed more extensive criteria than we had heretofore had. This was about eight or ten months ago, I think. And that is what is probably being used by Project

Management now as a baseline for making its findings.

Q Again, could you, for the record, just provide us with the identifying information and how we could find that reference?

A Yes.

Q However, let me go one more step with that question. You kept saying "technical capability." My question a moment ago dealt not only with the technical capability, but management capability and management procedures, including such things as some of our discussion earlier in this deposition about the matrix concept, project manager concept, and so forth. So when you say "technical" do you mean a pure technical capability or do you include management as I have tried to explain it?

A I meant pure technical.

Q Let me add another question, then. Does NRC, as a matter of requirement, impose any management format requirements upon utilities or major contractors?

A Maybe we do. Let me explain.

It is always felt that the quality assurance function of a utility should not be subservient to t .e construction function. So we would want a path of communications to a more senior official, perhaps a vice president. If that is management, then yes, we dabble in that area, yes.

Q Yes, that is exactly the aim of my question. Are there any other things like that that you can think of?

A Yes. We look for safety review boards, and require what is known as a plant audit by the Plant Operations Review Committee, a safety committee, and require certain qualifications and numbers to oversee the activities of a plant that changes, and so on.

In addition to that, there is an in-house safety committee.

Q Are these required in the SRP document or any other similar document? And I call your attention particularly to Chapter 17.

A They may be, but I haven't read Chapter 17 in so long I couldn't tell you. But I can find out.

Q Yes, would you, for the record.

Dr. Ross, I can't tell you how much I appreciate your responses today. And let me add one last question:

Do you have any observations you would care to make concerning safety engineering and management practices at NRC, or for that matter, at utilities or the major contractors?

A Well, in-house the agency is going to have to do a better job, as I have already stated elsewhere, tracking disturbances at plants and seeing that they do or don't have generic implications. The Davis-Besse event of 1977 was an outstanding example of how not to do something. And the same thing goes for the industry. They are going to have to track the same things we track.

But I guess that's all I have.

MR. MILLER: Thank you.

MR. PARLER: Mr. Allison, are you ready to proceed with your questions?

MR. ALLISON: Yes.

BY MR. ALLISON:

Q Dr. Ross, just a couple of short questions to follow up on some of the things Mr. Miller was discussing with you. I believe in response to one question you stated that the Office Director might override the decision made by the project manager and that the Commission could do that, could also override such a decision. I just want to ask you if, in your opinion, in fact the Commission itself is very much inhibited from doing things like that by the ex parte rule.

A We had an instance -- let's see if I can think of the details. The Commission is more involved since March 28 than it was before, and there were some decisions -- I can't think of it right off, but there were some decisions involving the restart of the Rancho Seco or the Davis-Besse facility, both of whom were under petition for rulemaking -- not for rulemaking, for hearing. The Commission, nonetheless, listened to the Office Director and took notice of the fact that they -- they were listening in an executive sense, and took note of the fact that in a judicial sense they might have to act on that matter later on.

I know that they have made some decisions recently on changes to plants, and I think in some instances they have given additional guidance beyond what the Office Director made, knowing full well that on one or more of those plants they may have to sit in a judicial sense.

I don't know if the ex parte infringes on them or holds them back or not. I just can't tell.

MR. PARLER: These recent events that you were talking about, Dr. Ross -- am I correct in my understanding that these matters that you referred to were discussed in an open Commission meeting? Isn't that right, sir?

THE WITNESS: That's right. And the Commission, when it issued a statement, clarified in advance the extent to which they wanted to take information from the staff. And I believe they served this on the parties, although I am not sure, to make it clear to the parties that they didn't think this violated ex parte.

MR. PARLER: Go ahead.

BY MR. ALLISON:

Q With regard to the relative technical strength of Metropolitan Edison relative to other utilities, have you had an opportunity recently to study the numbers of experts of different types that are available to Met Ed, including the GPUSC organization, in relation to those that are available to other utilities?

A No. I read through a recent submitt l -- recent in the last few days -- by Met Ed, relating to restart. And there was some material in there on it, but I didn't study it.

Q In your work at the plant site or later, after the accident, did you ever say or hold the opinion that the utility was technically think or weak in relation to either the needs of the accident, of the utility, or in relation to some other standard?

A I don't know if I said it or not, but I held the opinion for the first few days that they were weak. But it was an absolute determination, not weak compared to somebody, but weak for them.

Q It was not compared to another utility?

A Well, I don't think so.

Q Was it perhaps compared to the needs of the day?

A That is correct. In particular, a good amount of the technical work was being done at Lynchburg by Babcock & Wilcox for the utility, and Lynchburg is 400 miles away.

Let me make it clear. This is not to say that they didn't have the expertise available to them, but it was at B&W, not at Met Ed.

Q Dr. Ross, do you recall participating in a briefing of the Industrial Advisory Group on the evening of Sunday, April 1?

A Yes.

Q I think my first question about that meeting then would be to ask you to tell me what you recall of it, what its purpose was, and what happened.

A All right. Well, the arrangement -- I was more or less working nights, and working nights meant from 4:00 in the afternoon until about 10:00 the next morning. When I got to work Sunday afternoon, I found out that this group was assembling up in a National Guard barracks in Middletown, and they were due to be there about 5:00 or 5:30.

So I made some notes and organized a briefing that would be given to these gentlemen to get them up to date. Mr. Denton was going to go, but I believe he had to go to a press conference. Chairman Hendrie, I think, attended for a while, the first hour, and so did Roger Madsen.

We discussed the status of the facility and the events that had taken place to that time. There were a large number of people there, I don't know how many -- 30 or 40 -- people who I recognized as being the senior reactor safety people from around the country, from utilities, as well as consultants. We talked for an hour and described how the industry group could subdivide and assist.

I should mention that the utility was represented by Mr. DeCamp, the President of GPU. As I recall, there were three subgroups formulated, one under Mr. Zabrosky to look at the core damage, one under Mr. L vinson to look at, I think, recovery, system recovery. And I can't remember the third one.

I don't have my notes with me.

One of the things that grew out of that evening's meeting -- the NRC had called on a lot of national laboratories, and I was supposed to collate all the national laboratory work so we could prioritize it and avoid redundancy, gaps, and overlaps.

I think the meeting broke about 7:00 or 8:00 o'clock, and I went back. That's about it.

Q Do I understand correctly that your function was to brief the group on what was happening in the plant?

A That's right.

Q Did the utility also brief the group on that?

A On what?

Q Did Mr. DeCamp also brief the group?

A I don't think he had much to say, as I recall.

Q Do you know of any earlier briefings of that group?

A No. I'd be surprised if there were any.

Q You mentioned the formation of three subgroups. Do you recall any specific tasks that were given a high priority at that time?

A The purpose of the Zabrosky group was to try to formulate a core damage model to be used as input to the safety of going to cold shutdown. It was such a fast-moving target that what was decided on Sunday may have been overruled on Monday, and so on. But I recall that Mr. Levinson worked quite a bit on the plan to go to cold shutdown.

There are a lot of these notes that are part of a collection in two cardboard boxes over in Room P118, and they are organized in some fashion. I understand that John Collins had them shipped up a few days ago. I haven't had cause to go over there to go through them. Access is somewhat restricted. So I haven't reviewed what I did that Sunday evening, and my memory is a little fragile.

Q So do I understand that you don't have a set of notes from that meeting in your office?

A No, I don't.

Q But you believe there is material in P118 about it?

A I suspect there is. There is a filing system, but I am not familiar with it. I looked at it this morning and it looked to me fairly well organized, but I don't know who organized it.

Q I take it the basic function of this meeting was to brief people and get them up to speed, and then get them organized and off doing things; is that correct?

A That is correct. I do know that the fuel damage group was very active, and within a few days they had subsequent meetings in Bethesda, I believe it was on the following Wednesday or Thursday -- I think it was Thursday. And they also met in Lynchburg, and there are some very detailed minutes that were prepared. Ralph Meyer of BSS has a lot of this material. And they quickly formulated, based on the information they had, a model of the core damage.

I didn't work actively with that group following that night, so I don't really know what they did too well after that, except for this fuel damage group.

Q Did Dr. Matson or Dr. Hendrie participate actively in the briefing?

A Dr. Matson did. I believe the Chairman was more of an observer. He couldn't stay long. This was shortly after the President's visit, and I don't know what determined his activities for the rest of the day.

Q Did Dr. Matson brief the group on the hydrogen bubble?

A I believe he did; I believe he did.

Q What can you recall about what he said about it at that time?

A I don't recall. My recollection is that the bubble had diminished somewhat by then. No, I don't recall. If anyone took notes of what anyone said at the meeting, I am not aware of it. I know I didn't, and I know Matson didn't.

Q Do you recall anything else about bubble discussions at that meeting?

A No, I don't think so. Nothing comes to mind; no.

Q Did this group make any specific request for assistance or support from the NRC at that meeting?

A I don't know if they requested it, but I furnished it.

One of the people I had organized to go up was a senior Operations Licensing Branch examiner. His name is Jerry Holman. He is very familiar with the plant and procedures and people. And I made him available to that group for several days so they would know what the reactor would look like and what the pumps were, and so on.

Q Going into the next few days, then, were you familiar with the workings of the IAG?

A I didn't work with them too much. What happened is that about this same time there was a collection of utility executives from around the country coming in, notably Mr. Byron Lee from Commonwealth, and Bill Lee, President of Duke Power, and Fred Stern from Combustion Engineering, to assist the utility and senior management expertise. They formed various working groups and an ad hoc recovery organization. They met every morning, met every night, made decisions. And I worked more with them, because these were the people that were stationed at the site; they were making the decisions. The IAG became more of an oversight.

Q Do you know how the NRC was represented in the IAG?

A In the IAG?

Q Right.

A I know how it worked in the fuels damage group section. Billy Meyer and Mike Toquar were represented. That is the only one I am familiar with.

Q Okay.

Do you have an opinion on whether or not the IEG was an effective organization?

A By and large I don't think it contributed too much, except for the core damage model. I think that was very effective. But I think its usefulness was overtaken by the utility executive team I have described, which was very quickly converted to an ad hoc management structure, where people from all over the country were directing recovery work, not just the utility. The utility, I don't think, could have done it all on its own. And that became very effective.

Q Was the IAG possibly also useful as a backstop to the utility management team in thinking of issues, things like that?

A It may have been, but I didn't watch that interface. At that time there was concern about such things as: If you lost outside power, would the core melt? And if it melted, how long would it take to melt through? What would be the consequences?

And the IAG was doing some deep thinking like that, whereas the recovery team was worried more about putting in extra power lines so you wouldn't have to worry about loss of outside power, and designing and constructing and observing a heat removal system, and rewriting the procedures to make sure our contingencies were taken care of.

I believe the IAG's influence waned quite a bit after the disappearance of the bubble.

MR. ALLISON: That is all I have.

BY MR. PARLER:

Q Dr. Ross, Mr. Miller has already covered a lot of the broad territory that I was interested in by some of the questions he had. Therefore, I think it is appropriate to repeat to you now Mr. Miller's observations at the outset, which were that if you have already testified to something that was asked, and if you can make ready reference to that, please do not hesitate to do so. We are not trying, in other words, by design to get repetitive answers or to have the same territory covered more than once.

I have also read and studied your deposition of August 2, 1979 before a representative of the staff of the President's Commission on the accident at Three Mile Island.

Is my understanding correct that the August 2 transcript, with your corrections, and the accompanying exhibits to that transcript, is the substance of your appearance before the staff of that Commission? In other words, have you supplemented that material with material that was provided in the form of exhibits or material which in your judgment was significant?

A Before the deposition there was an informal meeting, and as a result of the meeting there was a request to get some documents, which I sent. I can't remember now what I sent, but there is a record of it. Because I sent a copy without the exhibits to Tom Rame. And I don't even remember what they were now.

Q But other than that, there is nothing else that stands out in your mind?

A No.

Q Now, before proceeding on a variety of items which may give the record the appearance of jumping from one area to another, I would like to start out at a point where I believe you and Mr. Miller left off a few minutes ago.

As I understood one of Mr. Miller's questions to you toward the end of his questioning, the substance of the question was what were your views about the strengths and the weaknesses of what I would call the regulatory program, the regulatory process. Mr. Miller perhaps used different words, but I sensed that he was talking about the same thing that I am asking you now, by your answer to him which he accepted.

And your answer was that there has to be a better system for tracking disturbances which have generic safety implications. You referred to the Davis-Besse event of September 24, 1977, and you said you would apply the same thing to the industry in its entirety.

Now, having said all of that, my question to you, sir, is: Do you have anything else to add on the major areas that you think need improvement -- and not limiting those areas just to the NRC, but to the industry as a whole.

Is my question clear?

A Yes.

Q All right.

A I think within the NRC we will have to do something on maintaining and improving the degree of understanding that the people have for how a reactor operates. I perceive that as the number of research and test reactors dwindles, as we hire new people, the only way we are going to get reactor expertise is either from the utilities or from the Naval Reactors Program. This source is limited. We are going to have to find a way to get the proper blend of experienced reactor people in the agency.

As far as the utility industry is concerned, there may be a need for some kind of a technical ombudsman of sorts to counteract the preoccupation with generating electricity. There may need to be an official whistle-blower. Some smaller utilities develop an extensive dependency on their plant. One nuclear plant might represent 40 per cent of their whole generating capacity. So it's a terrible economic decision for someone to make to shut the plant down. Some strengthening there probably is indicated.

When we get to the point where the utilities are more willing to shut themselves down and take some more of the burden off the agency, I think that will be a desirable degree of maturity.

I don't have anything specific, though.

Q Dr. Ross, continuing along the same line, what in your judgment do you believe are the strengths of the regulatory system that the NRC has which attempts should be made to recognize and to preserve if major changes are made in the system?

A I think there are several significant strengths. I would put integrity of the people at the top. I don't know where any allegation has ever been made, and certainly substantiated, anyway, that the people had less than 100 per cent integrity. I think we have succeeded in getting a high caliber technically as well as morally. I think the agency has enough money through its research program to bring a large number of qualified scientists to bear on a particular problem in a very short period of time.

I think the agency is independent, despite claims to the contrary. I don't think the agency hesitates to take an upopular action with respect to reactor restriction or shutdown if it is needed.

So I would put integrity, intellect, and independence as the strengths.

That's it.

Q Now, with regard to your deposition, Dr. Ross, of August 2 before a representative of the President's Commission, there are a couple of areas that I want to ask you questions about to make sure that I understand the transcript correctly. I want to ask the questions only for that purpose.

Do you have a copy of the transcript with you?

A Yes.

Q I believe that on page 18 -- and you would perhaps have to start out with your answer on the bottom of page 17 --

A Right.

Q -- when you are talking about levels of sensitivity that would bear on whether there is an information flow between certain divisions -- I guess the Division of Systems Safety and the Division of Operating Reactors. That was the context of what you were talking about. And then you said:

"There are exhibits and examples I could show of this information explaining this."

I wonder if you could provide an example, after you read what I was just trying to describe but perhaps did not describe clearly. After reading that, could you give me a typical example of what you were talking about there?

A Yes. In the emergency core cooling system for boiling water reactors, some of the pumps that were designed to pump water into the reactor, like any pump, have a potential to pump more water as the back pressure goes down. The less the friction pressure drop, the more water it pumps.

This is good up to a point, and then as you in effect start short-circuiting the pump, the pump goes into what is known as as a run-out condition, and it pumps too much water and it has a danger of undergoing cavitation, which is forming bubbles inside the pump and maybe hurting the pump itself.

DOR sent -- I don't have the number, of course, but DOR sent DSS an information report while they were reviewing some operating reactors and discovered that changes had to be made to prevent run-out, and that DSS might want to include this in their review of plants under construction.

Q Right.

A But I can get you copies.

Q Well, if it wouldn't be too much trouble --

A No.

Q -- a copy would be good.

Startir at about page 40, there was discussion with regard to whether you knew at that time whether the auxiliary feed water system for the TMI-2 plant was treated by the staff in its review as performing a safety-related function. Do you know what the answer to that question is now, any more than you did then?

A No, the information is in this document that was just filed a few days ago, I believe, but I haven't gone back and looked at it.

Q Don't bother. I just thought I would ask that.

Now, beginning at about page 45 through page 50, I believe questions were asked regarding transients in foreign reactors.

Is my impression correct that at the time of your testimony, because of things beyond your control such as understandings between governments, some of the information that you were talking about had not at that time been released to the public? Or do I have my transients confused?

A No. Let me make sure (examining document).

Q Just take your time, sir. On about page 45 there is a discussion of the coincident logic, and then starting about page 49, there was, I think, a discussion of a transient several years before in a foreign country.

A Yes; okay.

From August '74 until April of '79, the information about the transient wasn't distributed because the agency didn't know anything about it. We found out about it -- by the way, I brought with me a detailed chronology and bibliography of the whole thing.

Q That covers the situation after the decision was made by the authorities involved that the material could be released to the public?

A Yes. This is just a bibliography. Yes, it covers that. The references themselves are in a notebook about four inches thick. An Office of Inspection auditor is inspecting this also, so I prepared this for him.

We found out about it, as I recall, in late April. I think April 26 was the first the agency knew about it. And we were told at a meeting that we were having with Westinghouse.

As the material developed over the few days -- and I notice here that there were some telephone calls in May. I wasn't in on the phone calls. Mr.Tredony was, and Howard Faulkner. I don't have the date, but it is in early May.

Q All right.

A We were constrained by international agreement at that time from disseminating the information to the utilities. And I guess there has been no detailed information made available to the utilities formally yet.

Now, just recently, about four or five days ago, I signed a letter with an enclosure, transmitting the report on the matter, together with some of the minutes from Mr. Tredony to all power reactor operators. That may be the first formal transmittal.

Now, informally Westinghouse gave to the Presidential Commission a report, who promptly put it in the record, in effect declassifying it. So informally it was available about a month ago.

Q Is this formal report that you were talking about a voluminous document?

A It is about 40 pages -- wait a minute. I'm sorry; maybe I answered too soon.

Q The letter that I thought you said had been recently sent to the utilities --

A I signed the letter. The letter is only a paragraph long. It encloses the description of the transient which occurred at Beznau. The enclosure is about 40 pages long. It also enclosed a report on a recent transient at a facility in Belgium. Since I was sending it out, I just thought I'd send both of them. But the report itself is about 40 pages long.

Here it is. You're welcome to this copy, if you want.

Q Oh, may I have it?

A For completeness, let me show this to you also. That is the other enclosure that has gone out. It is a description of a steam generator tube rupture at the Doel reactor in Belgium. It has nothing to do with Beznau, but as long as I had it I just sent it out.

Q So the one on Beznau that was sent out was, I gather, the declassified September 4, 1974 report.

A That's right. That is supposed to be exactly what was received by the Presidential Commission. And since it is in the official records, I felt free to send it out.

Q This was --

A I don't think there is any difference. I don't think it has been expunged or anything.

Q As far as I am aware, that is the case. But this was sent out just with a letter of transmittal over your signature; is that right?

A That's right. I don't know whether the utilities have received it, but it has been mailed.

MR. PARLER: Even though this document that Dr. Ross just handed me is already in the public record, I think that it would be appropriate at this point to mark it for identification as Exhibit 1153. The exhibit is a technical report on the Beznau Unit 1 incident of August 20, 1974, TG-1 Trip, and it has a date of September 2, 1974 on the front.

(The document above referred to was marked for identification as Exhibit 1153.)

BY MR. PARLER:

Q Dr. Ross, are these extra copies?

A Yes.

MR. PARLER: Dr. Ross also handed me a copy of a memorandum from C. J. Heltemes, Jr. to all Bulletins and Orders personnel, forwarding a report on an incident at the Belgium Doel 2 reactor. I will mark that as Exhibit 1154 for identification.

(The document above referred to was marked for identification as Exhibit 1154.)

BY MR. PARLER:

Q So the discussion that you were having with the questioner on August 2 in the area that I previously referred to, on pages 49 through 50 of the transcript, was about the Beznau incident of September 2, 1974?

A That's right.

Q All right. On page 65, Dr. Ross, again at the top of the page but you will have to go back to the preceding page -- the context here is communications between the Division of Systems Safety and the Division of Operating Reactors.

As far as the substantive matter that is concerned, I have no question about that. My question is with regard to your reference to a written agreement between the two divisions, presumably dealing with the flow of information.

I wonder if you could be a little more specific on what that agreement is. And I will be more specific in my question.

A I understand. At the time the principal engineer working on this problem was in DSS.

Q Right.

A The Division of Operating Reactors had definitely a safety concern, and I believe that there was an agreement on lead responsibility.

Q Just for that particular thing?

A Yes.

Q I asked the question because I thought possibly you might be referring to some sort of interface agreement between DOR and DSS such as the one that exists between NRR and I&E.

A No, this would have been an ad hoc agreement.

Q Right.

On page 74, please, sir, about the bottom third of the page, the question was asked whether when they do conduct a review of each piece of equipment, safety-related equipment, it is done on a piece-by-piece basis or it is done on a more general basis. And it is your answer that I want to ask you the question on. Your answer is, "It is done on a piece-by-piece basis. We don't audit that."

Now, I gather from that response that you were saying that the audit mode of review--the licensing review that the staff generally operates in sometimes for certain things, shifts to some more detailed thing, an audit review.

A Yes. The reviewer might decide, for example -- and this is what I mean by "audit" -- that he is going to look at a pump that is used to recirculate water. He might ask for a test on the pump, the detailed pump characteristic, the quality of the pump in the accident environment, how the operators would align the pump to be used, what would happen if the valve that let water in the inlet were inadvertently closed for 30 seconds, how the pump works without any cooling water to the bearings. And he never asked this question before, or he only asked it on every third application, or whatever.

That is what I meant by an audit. When he looks at something on an audit basis, he looks at it in detail, but he doesn't look at the whole plant in that detail.

Q Bear with me for a minute while I go through these other notes (examining documents).

The enxt question that I have is on 98. Oh, excuse me; I have one on page 97 first.

At the top of the page there was reference in a specific context to the systems interactions from branch to branch. The question that I am going to ask you really has nothing to do, I don't believe, with the context of your answer, but the reference to the words "systems interaction" suggested to me that I ask you whether, during the months that you had occasion to serve as the Deptuty Director of the Division of Project Management before you were relieved of that responsibility, I gather, in March to perform TMI-related work, did you have occasion to get involved in what your staff was doing under Project A-17, the systems interaction study?

A Yes. When I took the deputy director job, with it came the function of task supervisor of that item. The task manager was John Angelo.

In connection with that responsibility, there were several meetings that were held with Sandia and with the AIF advisory group on it. Sandia is our contractor, and I went to Albequerque twice, I think. We met with an ACRS subcommittee to describe what we were doing, and we met with the industry group. So yes, that was it.

Q Now, the next point is a follow-up point, that it would appear to me from reading some things -- and we also deposed Mr. Angelo -- that there may be a fairly substantial difference of opinion as to what the ACRS wants and what others feel is realistic to accomplish.

Was that your impression?

A I think so. I wanted to parcel the work out and get things done one step at a time, rather than study things for years and never get anything done.

So what I did was focus on a part that was doable, which was to say what the safety implications are during the cool-down of a plant from when it is operating to when it is down on what is known as its cold shutdown equipment.

And we identified three bad things that could happen during that. You had inability to shut the reactor down, inability to keep the pressure down, and inability to remove decay heat.

So we looked at the way systems could interact to produce one or more of those three undesirable things.

I don't think we had any substantial difference in the committee as to what should be done, but it was when and how. And my interest was getting a number of short-term things done, each with an end.

Q Now, going to page 98, about the middle of the page, your response to a question as to whether your Bulletins and Orders Task Force was going to be an ongoing thing, my question goes to your response. I think that I understand it, but would you explain it to me, please, sir.

Do you see what I am talking about?

A Yes. Task forces in general, I think, reveal weak organizations.

Q That is why I asked the question, to make certain that I understood that is what you were saying.

A That's right. And I would like to dissolve it and get back to where we belong. I think they are all right for a short period of time, but if you lean on them for a long period of time, it just means you have a bad organization.

Q On page 101, six lines from the bottom, you say that since the TMI-2 incident, your task force, the Bulletins and Orders Task Force, has worked a lot with operator training and operator licensing. And although my question does not relate directly to what you are responsible for, I wonder: Are both your task force and the Lessons Learned Task Force looking at operator training and licensing?

A Yes. They are looking at it from a broad viewpoint, and I am looking at it from a narrow viewpoint.

Q "Narrow" is more immediate attention?

A Narrow, and also limited in scope. My task force looks at the loss of water, loss of feed water events, and the operator training procedure associated with those events.

Q And is my understanding correct that before March 28, 1979, generally speaking at least, the regulatory agency did not review operating procedures in detail and make the necessary correlation between those procedures and the safety analysis review?

A That is correct.

Q Okay.

A The two disciplines that did look at them were the Operator Licensing Branch, because they wanted to give oral and written exam questions to the operators on procedures, and the Inspection and Enforcement, who among other things wanted to make sure that the procedures didn't violate the license.

Q And one of the things your Bulletins and Orders Task Force is doing is supplying that need, I gather, on an ad hoc or temporary basis?

A Yes, but that is only a small part. The main part is to see that the technical analysis of the reactor suppliers is translated into language the operator can understand.

Q You mean where these procedures really originate?

A That's right.

Q Okay.

On page 106, about the middle of the page, you talked about developing a plan for verifying plant transient predictions in a detailed manner during startup tests. Is my impression correct that prior to March 28, 1979 -- and again this is a general statement -- the startup tests at a nuclear power reactor were witnessed, if at all, only by people from the regional office of

I&E?

The question should be sharpened up a little bit. In other words, prior to March 28, 1979, generally speaking there wasn't an NRC team that witnessed these startup tests, was there?

A Did you mean an NRR team?

Q Yes, an NRR/I&E team is what I mean.

A That's right. Does your deposition copy have "years" instead of "weeks" there?

Q Yes. And I focused that on the second reading, and I guess that is why I am asking the question. I think the answer to the question I have already asked has revealed what was kind of puzzling to me. Until I saw the "years" there for the "weeks," I thought the plan, generally speaking, was just for the regional inspectors to witness some of the startup tests.

Now, I gather from what you have said here that there is a more ambitious plan than that, that was started being developed about two years ago; is that right?

A Yes. It is not in the deposition, so let me clarify that.

Q Okay.

A The Analysis Branch in Reactor Safety has a responsibility for reviewing and approving the plant transient methods. The Analysis Branch didn't exist until January '76. During the first year, when we were trying to develop the mission of the branch -- and one of the reasons the branch was created was to review and approve some plant transient methods that had never been reviewed and approved -- we observed fairly early that we didn't have the plant data that would be used to validate the methods. So we developed a plan to get the data, and that is the plan that is referred to here.

What we decided on was that we would pick two or three plant transients and make sure they were heavily instrumented, and during the startup test the appropriate data would be gathered, compared with the analysis done by the supplier, and then --

Q So it was a plan to get more data, not what I thought, the NRC team that would have been there to see the test.

A That's right. This whole test could have been done without ever going to the plant.

Q Fine. I'm glad I asked that question.

Let me ask you this: From page 108, at the bottom of the page, I gather at your deposition on August 2 you only had certain notes with you, up through a certain date. Do you see what I am talking about at the bottom of the page?

A Yes.

Q My question is: Have you found out anything else since that time which sheds any different light on what you were saying?

A Yes and no.

Q Okay.

A As far as my diary is concerned, my recollection then and now is that I had taken the pages out and turned it over to my replacement who said he may have thrown it away if he didn't see any need for it.

But there was a memorandum that existed that I had signed that eluded my memory at this deposition, and it was provided to me later. It is the October 19, 1977 memo from me to Karl Seyfrit.

Q Is this what you are talking about (indicating document)?

A Yes -- October 20, pardon me. Yes, that is correct. I had forgotten, during the conduct of the August 2 deposition, that I had written this memo.

Q Why don't you hold that one, Dr. Ross, for a second, and let me digress from the transcript of your deposition to cover some points in that memorandum.

That is the memorandum that you had in mind, right, the thing I just handed to you, from D. F. Ross to Karl V. Seyfrit, Assistant Director, Division of Reactor Operations Inspection, IE. It is dated October 20, 1977. The subject is: "DAVIS-BESSE ABNORMAL OCCURRENCE (9/24/77)."

A Yes, that is what I was referring to.

MR. PARLER: I will mark that for identification as Exhibit 1155.

(The document above referred to
was marked for identification as
Exhibit 1155.)

BY MR. PARLER:

Q Some of my colleagues wanted me to ask you a couple questions about this memorandum, the first of which you have already answered. Did you write a note dated October 20, 1977, in which you raised some areas of interest --

A Yes.

Q The rest of the question is: -- concerning the the Davis-Besse incident of September 24, 1977? The answer to that is yes, you wrote that memorandum.

Was this memorandum part of your normal job-related function at that time?

A Yes.

Q The next question is self-evident: To whom did you send the memo? Obviously, to Mr. Seyfrit.

All right, why to Mr. Seyfrit?

A It is my recollection that as a result of a meeting in Dr. Mattson's office in early October -- and I don't recall the date -- it was decided that I&E would be the principal spokesman for the NRC with respect to followup on this occurrence.

Since I had sent one of the people in Reactor Safety, and since he brought back information, I thought it important to make sure that these very points were factored into the I&E investigation.

Q That was Mr. Mazetis?

A That is correct.

Q So this was a disposition memo that would --

A In effect we were saying, "These are the areas we would look into if we were doing it, and since you are doing it in I&E, this is what we think you ought to be doing."

Q I gather that it was not sent directly to the region because the normal communication channel at headquarters on something like this is with I&E headquarters; is that correct?

A That is correct.

Q I realize that you aren't the appropriate person or the best person to ask this question, that it should be addressed to the I&E headquarters recipient, but do you have any knowledge as to what was done with this memorandum, if it was sent to the regional office?

A No, I don't.

Q All right.

A I do know -- I later found out, sometime this spring, that there was an Inspection and Enforcement report prepared, but to the best of my recollection I did not get a copy.

Q Of the report that was prepared?

A That's right. I may now have it. I think I have a copy of it.

Q Oh, I am talking about before March 1979.

A No, I know I didn't get it then.

Q Do you know if your memorandum was ever sent to the inspectors that were investigating the incident?

A I don't have any knowledge.

MR. PARLER: All right.

Off the record.

(Discussion off the record.)

BY MR. PARLER:

Q If we can go back to the transcript of your deposition before the President's Commission, let me see if I have any other questions about it.

On page 131, your answer starting at line 13 and continuing through line 16, in the interest of clarity could you tell me who it is you are talking about there? In other words, who is the "they"?

A Oh, B&W Design.

Q Right; I thought that, but just out of an abundance of caution, without being a nit-picker, I thought I'd ask the question.

B&W still doesn't have as many secondary side reactor trips as Westinghouse does?

A The same today, on line 5.

Q I thought I'd better clarify that, because the impression that some have who are not as familiar as you are with this thing, is that they do have more.

A I understand.

Q On page 135, at the top of the page, lines 6 to 9, "It would have pointed out the role the integrated control system plays in these transients, and that would have been fixed."

I gather -- and this question is not necessarily in the nature of a clarification of what you said there, but in the nature of an elaboration of what was said there.

It is my impression that control systems generally, including the integrated control system, received little or no regulatory review prior to March 28, 1979. Is that a fair assumption?

A Yes, that is correct.

Q Now, again for possible completeness on this particular point, in general terms could you state how the integrated control system for a B&W plant plays an adverse role in these transients -- not a technical treatise, but one that would give the highlights to a layman.

A Yes. The integrated control system, among other things, controls the feed water flow, and an adverse role would be that at the time you needed more feed water, it could produce less.

Q All right. That's it for going through the transcript. I'll put that aside, except I have one question, not about the transcript but about one of your exhibits. And I don't think you'll have to refer to it, because I just want to relate it to something.

It is your April 25, 1979 NRR status report on feed water transients in B&W plants.

A Yes.

Q I am having a little difficulty distinguishing that from or relating that to the Tedesco Report, NUREG 0560. Is the April 25 thing an earlier version of that NUREG, or what?

A It is an earlier version. Some of the material in the April 25 report was taken from the Tedesco Report. The purpose was to get enough material to advise the Commission of action to be taken on the B&W plants.

MR. PARLER: Why don't you ask Dr. Ross your questions now.

BY MR. COX:

Q Dr. Ross, you addressed a little while ago your understanding of the word "audit" with respect to what a reviewer really does in at least one sense. Could you give us a few more thoughts on that, on your understanding of the word "audit" as applied to how a reviewer uses a standard review plan on a given project assignment?

A Okay. I used "audit" in the sense that we don't do a detailed design review. If we did, we would need maybe as many people to do the design review as did the design, which would mean thousands of people. So we don't review everything that is done.

Now, the standard review plan, if followed, should -- let me start over.

The reviewer is not supposed to audit in the sense that he does some of the standard review plan but not the rest. That is not what I meant when I said "an audit." If the standard review plan says, "Review these 23 things," then that is what he should do. But he may decide, in terms of depth on any one item, that he is going to go a lot deeper on an audit basis than he did on the last plant, or than anyone else has ever done.

That is what I meant when I said "audit."

Q Then, with regard to section 2 of the Review Plan, it lists acceptance criteria in each of the standard review plans.

A Yes.

Q There is a section that says "Acceptance Criteria." Does the reviewer then check projects, submitted material, against each of those acceptance criteria?

A He should.

Q He is supposed to?

A He should.

Q Regarding the NRR/IE interface, do you believe that the memo to L. V. Gossick from B. Rusche and E. Vogeneau, dated March 21, 1977, subject "AGREEMENT ON NRR/IE INTERFACE AND DIVISIONAL RESPONSIBILITY" -- are you familiar with that?

A I may have read it, but I don't remember.

Q It was recently reissued by Mr. Denton this summer as a reminder.

MR. PARLER: Let him take time to look at that for a second.

THE WITNESS: I remember reading this when it came out, and I remember Harold's recent admonition.

BY MR. COX:

Q Let me ask it this way: Given the current understanding that you feel is held by all the personnel involved in the NRR/I&E interface, and your understanding of how it is implemented in the process, do you feel that a comprehensive evaluation of licensee performance and adequate feedback to NRR is reasonably assured from the way we handle this interface?

A I am going to have to -- I think I just have to say by and large I haven't been involved with that aspect. I never worked in DOR and that is the principal agent that interacts with I&E. So the only thing that I have done is through the Task Force. I have been working very closely with I&E, but I don't think I could generalize.

Q If I could just pursue that for one or two more --

MR. PARLER: Go ahead.

BY MR. COX:

Q There is a time generally when a reactor first becomes licensed to operate that the project still stays in DPM.

A That's right, a few months. It has been as long as a year, and one notorious example of Fort St. James, several years, still not transferred to DOR.

Q In that case wouldn't the interface be more or less between DPM and I&E?

A Yes. For the few months of interaction, there is a parallel project manager from DOR assigned to the project in an unofficial capacity, even before it gets transferred. TMI-2 was a good example. I don't believe TMI-2 was transferred. Yes, TMI-2 had not been formally transferred as of March 28 this year. Nevertheless, DOR picked up immediately as the primary agent.

MR. PARLER: You mean even before the transfer?

THE WITNESS: No, from the time the event happened.

MR. PARLER: Oh, yes.

I just want to show you this. You have already said that in your past experiences you didn't get involved too much in the DOR/I&E interface; isn't that right?

THE WITNESS: That's right.

MR. PARLER: But this DOR Memorandum No. 2, which I have just shown you, does clearly state that it is the responsibility of the Division of Operating Reactors to continuously assess the pertinence of information obtained from operating reactors, and that significant findings are to be forwarded to the appropriate division in a timely and informative manner.

That is what it says in the second paragraph. It doesn't seem to make any distinction between where the lead responsibility is.

THE WITNESS: I think in context, though, this did not apply to the reactors in the first few months of operation.

MR. PARLER: Yes.

THE WITNESS: But I have developed a more detailed appreciation in the last few months from the Task Force, because all I have been working with is operating reactors. And I have developed a heightened appreciation for the NRR/I&E interface. However, I have lost what the question was.

BY MR. COX:

Q Based on that heightened appreciation, then, maybe you could answer, or you might want to make an observation on my next question. If you don't feel that you want to, just let us know.

A All right.

Q Do you feel that I&E, then, has the technical capability, both in type and quantity, to really identify and inform NRR on a continuing basis of potential unresolved safety issues, whether they be procedural or design oriented from the field?

A I think they have it, but largely because I feel like NRR hasn't, let's say, spread the gospel enough. I don't think it is being used.

Let me explain. This is from my heightened appreciation from the last few months.

The title of the task force is Bulletins and Orders, and we have issued or created about eight or ten bulletins. One of the last bulletins -- in fact, the last bulletin set that we issued -- required licensees to do some things with their reactor protection system, emergency core cooling system, that were very puzzling to the I&E inspectors. They didn't have any idea why we were requiring this. So I and one of the people on the task force went to Region 4 and briefed the inspectors from Sections 3, 4, and 5, and then last week had a briefing in Region 2. I didn't go, but I sent somebody.

And from those two meetings I drew the conclusion -- and I am so reporting to the Commission -- that NRR has got to do a better job explaining technical policy to the inspectors, so that they in turn can do their job better about reporting back to us on the procedural designs you referred to.

I don't fault the inspector for not doing things. I think he could do a better job if we could just tell him what we are looking for.

And that will be the substance of my report to the Commission.

MR. COX: Thank you.

MR. PARLER: That is going to be in writing, do you think? I am not going to ask you to provide it, but I want to be able to look out for it.

THE WITNESS: That will be a position paper.

MR. PARLER: A position paper?

THE WITNESS: Yes. It should be on Harold's desk today.

BY MR. COX:

Q On another topic, emergency actions by the control room operators, specifically with regard to turning off ECCS, there are a number of background documents. I would just like to quote three for you that in one form or another intimate that perhaps operators should not be allowed, at least for some amount of time, degrading ECCS flow after it is automatically initiated.

One of these, and perhaps the latest, is NUREG 0600, the I&E report on the TMI investigation, that was issued in August 1979. On page 8 it says, "The throttling of HPI was one of the four actions that contributed to the accident."

Another document is NUREG 0138 of November 1976, entitled "Staff Discussion of 15 Technical Issues." On pages 4.1 through 4.11 it discusses a fairly complex issue, but I will paraphrase the staff position that came out of that. The staff position was to procedurally prohibit ECCS cutoff prior to ten minutes after automatic initiation.

A third thing that is more recent was a memorandum from

B. Dunn to J. Taylor, both members of the B&W organization, dated February 16, 1978. The subject was "Operator Interruption of High-Pressure Injection." And this document, by the way, is Exhibit No. 4 to Dunn's testimony at the Kemeny Commission.

MR. PARLER: Are you familiar with that?

THE WITNESS: I have that.

MR. PARLER: Are you going to ask questions about it?

MR. COX: Yes.

BY MR. COX:

Q My question is: Given all that has gone over the boards 'til this point, and with your detailed technical experience in this area, do you feel that there should be some minimum elapsed time after initiation of ECCS, during which an operator could not, either by procedure or by design, terminate the flow?

A I don't think it is technically feasible to do by design. In other words, I don't think there is a design that would preclude not doing something. If the operator is determined to turn off some pumps, no design in the world is going to preclude it. That is just literally impossible, except -- well, if you have armed guards stationed at the circuit breakers with instructions to shoot somebody who came close to it, that might do it.

Q Could I just interrupt for a moment. I meant permanent control.

A Well, ·gain the operators are very clever. You can take the button and put in a timer that says that if you've had an emergency core cooling system initiation, for the first ten minutes this button won't work.

He can defeat the design. Man can defeat the machine.

You used the term "could." I think in terms of "should" -- I don't think the operator should have to do anything in the first eight to ten minutes to assure core protection. He shouldn't be burdened with things he has to do.

As far as the time element is concerned, if the emergency core cooling system came on when it shouldn't and the operator has enough technical information to determine that it is the actuation of a spurious, then I see nothing wrong with his overriding it and turning it off.

Our task force is developing criteria for doing just that. These would be permissive criteria. And they would be based on the state of the reactor: Is it cool enough? Is it controlled? If you can determine it is spurious, then we would permit them to turn it off.

Q And this would include the case where he perhaps would be able to determine that it is spurious in a minute or less, if he could? If he could do that, it would be all right?

A In theory, yes. The termination would probably take longer, but if he met the termination criteria, then he could terminate it.

MR. COX: That's all I have.

BY MR. PARLER:

Q Are you aware, Dr. Ross, of any internal memorandum from the Division of Project Management to the Division of Operating Reactors which urges that the responsibility for operating plants should be transferred to DOR before the plant reaches an appreciable power level?

I realize that isn't in the mainstream of what has been occupying your attention since the end of March, but I am trying to find out if, in your capacity as Deputy Director of DPM, you were aware of such a memorandum.

A No, I am not.

Q Okay. Do you believe that internal procedures for effecting the transfer of projects from DPM to DOR are adequate?

A No, they are not.

MR. PARLER: Do you have any other questions?

MR. COX: Off the record.

(Discussion off the record.)

MR. PARLER: Back on the record.

BY MR. PARLER:

Q Dr. Ross, what has been the staff position or practice, to the best of your knowledge, with regard to giving credit for non-safety-grade equipment, such as pressurized relief valves, feed water control systems, turbine stop valves, pressurized heaters, et cetera, to mitigate transients and accidents?

A I am not sure that the premise of the question is correct with the "such as's."

Q All right.

A If we can limit it to just the use of non-safety-grade equipment --

Q Why don't you so proceed in your answer.

A The general practice in a transient accident is to assume that the non-safety-grade equipment is neutral; it doesn't help and it doesn't hurt. As far as safety-grade equipment, the most damaging failure in safety-grade equipment is assumed.

I think that's it.

MR. PARLER: Off the record, please.

(Discussion off the record.)

MR. PARLER: Let's go back on the record.

While we were off the record, I provided Dr. Ross with certain background documents. I don't know whether he is going to refer to them or not, but if he does, they will be so identified.

BY MR. PARLER:

Q Why don't you proceed, Dr. Ross.

A Okay. What I had indicated for design transients -- it is generally assumed that the control systems for non-safety-grade equipment items don't fail in such a way as to aggravate the event, and don't operate in such a way as to improve the event.

The staff policy -- I don't have it with me -- to document what I just said is being developed and written down in a clairvoyant fashion as late as yesterday.

We had spoken to this issue once before, and NUREG 1308, which as been previously referred to, or 0153, which is a sister report, I forget which one. Several years ago, as a result of recent disclosures by Westinghouse, it was considered necessary to reformulate and reissue our non-safety-grade equipment report to mitigate transients. That job was recently assigned, meaning Monday of this week, to Paul Check, who is a branch chief. Yesterday I read a draft and returned the draft to him this morning with the comment that I think it represents the right tone, and a few comments on how to fix it up.

Q What was the Westinghouse experience you were referring to?

A They made a notification to Public Service Electric and Gas with respect to the Salem 1 facility -- early September. As a result of that, we had a round of meetings with the regulated industry in mid-September. The minutes of those meetings have been written by my project manager who orchestrated the meetings, and they should be out on the street now.

The complete subject of those meetings was the use of valve safety-grade equipment.

I think the question on looking at these two documents might be if one document contributed to the other. The thing I don't know for sure -- well, let me explain the function. When the plant is shut down, certain events might happen that would tend to fill the pressurizer, creating the potential for an over-pressurization event. These events aren't generally classified as transients. A transient usually starts with the system in the hot operating condition.

The protection for the over-pressurization event, according to the document which is dated February 6, 1978 that I signed and sent to Dominic Vassallo, concerns the use of a pressurizer level as producing an alarm. And the information I don't have with me is: Is that a non-safety issue? So I can't respond as to whether there is a paradox or not.

The same statement would have to do with the pressurizer reliev valve. The function of that is important to over-pressure. I don't know if the relief valve is a safety-grade quality.

I do want to point out in context the issue as to whether these instruments or valves are non-safety grade has to do with their surviving a hostile atmosphere that might be created. The over-pressurization event does not have the potential for producing that hostile atmosphere, because the reactor is cold, or else there wouldn't be a concern in the first place. And when the reactor is cold you don't produce steam, and it produces a hostile environment. And it would take more study to sort that out. I just can't do it here.

Q All right. Would you have any objection if I would mark these things just for identification and put them in the back of the transcript?

A Oh, of course not.

MR. PARLER: The document that Dr. Ross has referred to, or one of them, the memorandum from him to Mr. Vassallo, dated February 6, 1978, will be marked for identification as Exhibit 1156.

(The document above referred to was marked for identification as Exhibit 1156.)

MR. PARLER: There is a memorandum from Mr. Vassallo to Edward S. Christenbury, dated March 29, 1979, which will be marked for identification as Exhibit 1157.

(The document above referred to was marked for identification as Exhibit 1157.)

MR. PARLER: Off the record.

(Discussion off the record.)

MR. PARLER: Back on the record.

BY MR. PARLER:

Q Dr. Ross, I handed you, when we were off the record, a note from yourself to Mr. D. Eisenhut, dated May 20, 1977.

A Yes.

MR. PARLER: I will mark that for identification as Exhibit 1158.

(The document above referred to was marked for identification as Exhibit 1158.)

BY MR. PARLER:

Q The note appears to raise certain questions regarding the organizational roles of the Division of Systems Safety and the Division of Operating Reactors. It is also my understanding that that is a subject that was discussed at some length in December of 1977 at, I guess, a DOR/ DSS retreat.

With that background, the question that I want to ask you is: Are these concerns that were addressed in your note to Mr. Eisenhut resolved, or are they still concerns as far as you know, or what has happened?

A Nothing. If anything, the situation is worse.

Q Could you elaborate a little bit, please.

A The context of the memo is that in many technical areas there are two groups working where, in my opinion, one is enough. And this produced a wasting of manpower as well as generation of diverse viewpoints on the same subject.

I had recommended that in any given technical area we only have one technical group. In Texas the motto would be, "One riot, one ranger," which is what they use for the Texas Rangers. That is what I was recommending.

If anything, since the time this was written two years ago, the situa tion has worsened, not bettered. We still have two technical organizations where only one is needed.

MR. PARLER: Off the record.

(Discussion off the record.)

MR. PARLER: Back on the record.

BY MR. PARLER:

Q Dr. Ross, did you ever get any response in writing or otherwise to your note of May 20, 1977 to Mr. Eisenhut?

A There was no response in writing. I understand that Mr. Eisenhut wanted to write a response and was told not to, words to the effect that, "This type of memo is divisive in nature and there is no need to exacerbate the division."

My hearsay information is that that advice came from Mr. Case.

Q So nothing happened as far as the concerns that were expressed in the note, and indeed you say the situation has gotten worse, that we in effect have two technical organizations?

A That's right. It is my opinion, based on discussions with Mr. Denton, that he intends to rectify this situation.

Q Okay. Incidentally, in your judgment, are there duplicate or competing centers of excellence other places in the organization that have some bearing on the safety function that we perform as far as nuclear power reactors are concerned?

A To a small degree. In the Office of Research there is a Fuel Behavior Branch, an Analysis Development Branch, and a Separate Effects Branch.

The Analysis Branch in Research is supposed to develop computer codes so that the NRR people will have independent analysis capability.

The Analysis Branch in DSS is supposed to apply the computer codes. Sometimes the interface between development and application becomes fuzzy, and the two branches kind of compete with each other. But it is not serious. In fact, that much competition is probably healthy.

MR. PARLER: Let's go off the record for a second.

(Discussion off the record.)

MR. PARLER: Back on the record.

While we were off the record I handed Dr. Ross some documents for him to examine. The first document is a note from D. F. Bunch to Dr. Ross dated May 18, 1979. I will mark that document for identification as Exhibit 1159.

(The document above referred to was marked for identification as Exhibit 1159.)

BY MR. PARLER:

Q Dr. Ross, this is a note which said that Mr. Basdekas called to express a concern that inadequate attention was being given to reviews of the control systems of PWRs and their effect on plant thermohydraulic stability.

Would you comment on that note, please, sir.

A Well, if you will notice, in the last sentence it says:

"The actions taken in response to the TMI bulletins do not adequately address this area."

No, they don't, because the bulletins had to do with the sequence of events that occurred at TMI 2 as they might appear to other PWRs. There were no thermohydraulic stability symptoms at TMI 2. So the bulletins shouldn't have addressed it.

The general subject of stability is a requirement in section 4.4 of the Standard Review Plan, and stability methods are reviewed by the Analysis Branch. If there is an adverse impact to the control system, they should pick it up.

In the course of my business as Director of the Bulletins and Orders Task Force, there is nothing I should have done with this memo, and I did nothing. It may be Mr. Tedesco, who now has the job that I used to have, may have done something with it, or he may be planning to do something with it, but I did nothing.

MR. PARLER: All right. Another document that I handed Dr. Ross is a memorandum for Mr. Denton from Dr. Ross, subject, "CONCERNS OF R. McDERMOTT." This document has a cover memorandum dated May 17, 1979, and it has attached to it, I believe, four references.

I will mark this document for identification as Exhibit 1160.

(The document above referred to was marked for identification as Exhibit 1160.)

BY MR. PARLER:

Q Dr. Ross, since this document appears to be related to your work as head of the Task Force on Bulletins and Orders, would you comment generally on what is involved, please, sir.

A Surely. There is some background material which is not written down.

Q All right.

A During the month of April, as the bulletin responses from Bulletins 7905 and 7906 came in, there was a working group within the task force assigned to review bulletin responses. Steven Garber was the head of this working group, and Bob McDermott, who normally works in the Quality Assurance and Operations Branch, was assigned to this group. I don't even recall now who assigned him. I don't think I did. Things were moving pretty fast in those days.

During the course of this, and especially after the revalation of what is now referred to as the Michelson Report, Bob McDermott's concerns expanded into areas not ordinarily associated with his responsibilities. He was concerned with the symptoms of the Michelson Report, the reliability of auxiliary feed water. I had other people working on these matter-, perhaps unbeknownst to him.

He wrote many memoranda which are not included here. There is a complete file available.

It was obvious to me that he was going beyond the scope of his job assignment. I felt it important to send to him the May 8 memorandum to get him to clarify in writing what his problems were. Not in the memorandum, but during my oral discussions, I tried to point out to him that other people, more qualified to understand the Michelson Report, were reviewing it.

I also asked his management, his branch chief and his assistant director, to be on the -- not the concurrence but the routing list so when he reported back to me it would be through his management, and I wanted his management to comment on his concerns also.

Mr. McDermott's response was on May 14. I think that is the second memo.

Q Right.

A And the other memos follow.

Q Right.

A I did ask Mr. McDermott, shortly after I went on an extended leave -- we briefed the Commission on the subject of the Oconee report, and on the telephone I offered to

Mr. McDermott the opportunity to express any contrasting points of view, and he declined.

Q Fine. The third and final document that I showed you, Dr. Ross, is a memorandum from F. W. Williams, Jr.

A Yes.

MR. . .RLER: This is a memorandum from F. J. Williams, Jr., who is a Technical Coordinator, Division of Project Management, to Darrell G. Eisenhut, Deputy Director, Division of Operating Reactors. The memorandum is dated May 17, 1979, and the subject is: "CONCERNS RELATED TO TMI-2 EVENT AND BULLETIN 79-05A - DON QUICK (IE: REGION II)."

I am going to mark that document for identification as Exhibit 1161.

(The document above referred to was marked for identification as Exhibit 1161.)

BY MR. PARLER:

Q In your capacity as head of the Task Force on Bulletins and Orders,were you aware of Mr. Williams' concerns, and if so, maybe you can comment generally on what those concerns were.

A Yes, I was aware of these. I met with Mr. Quick privately and discussed them with him. He is a regional inspector in Region II. I have high regard for his expertise. We used his resources when we were going to shut down B&W plants.

We went into each plant and discussed the training with the operators. We audited about 30 or 40 per cent of the operators at each plant for their understanding of the TMI sequence, their understanding of the new procedures for loss of coolant and for loss of all feedwater. And I think his advice was very helpful.

Q As long as we are dealing with documents, Dr. Ross, I want to give you two others which deal with a subject that we discussed some time ago, the foreign incident, the Beznau incident. I don't want to go back over any of the discussion, but the documents that I have given to you are a memorandum for the files from R. L. Tedesco, dated April 10, 1979, Subject: "WESTINGHOUSE ACTION ON TIM-2 INCIDENT."

As far as you are aware, is this the document that perhaps reflects the Westinghouse Corporation's notification of the event to the NRC? Or is this dealing with something else? That has never been clear to me. Thatis why I brought these documents up.

A It shouldn't have been clear to you, because it wasn't clear to me, either.

Q All right.

A At the time we got this letter -- and we took action on it within a week -- it was our understanding that Westinghouse wrote the letter dated April 10 from Tom Anderson because of the phenomena observed at Three Mile Island, not because of the phenomena observed at the Beznau facility. That is what we thought at the time, and I never had any reason to change my mind.

MR. PARLER: So that will be understood in the context in which Dr. Ross just put it, I would like to mark the document that we have been talking about, Mr. Tedesco's memorandum for the files, dated April 10, 1979, for identification as Exhibit 1162.

(The document above referred to was marked for identification as Exhibit 1162.)

MR. PARLER: The other thing I handed to you, Dr. Ross, was a memorandum from you to Mr. Case, who is the Deputy Director of the Office of Nuclear Reactor Regulation, and the subject is: "MEMO, DEYOUNG TO DENTON "PRECURSOR EVENT IN A FOREIGN REACTOR" DATED 7/24/79." The memorandum from Dr. Ross is dated July 27, 1979.

I will mark this for identification as Exhibit 1163.

(The document above referred to was marked for identification as Exhibit 1163.)

BY MR. PARLER:

Q Now, Dr. Ross, I handed you your memorandum to ask you a question about the next-to-the-last paragraph. The sentence to which I refer reads:

"The subject memo does not accurately represent Westinghouse plants as currently configured."

Now, what I wanted to ask you is: I gather that the current configuration of the Westinghouse plants which was not accurately portrayed or represented in Mr. DeYoung's memorandum, was due to some change that occurred in those plants after March 28, 1979.

A Yes, sir. "Currently" means any time on or after the 13th of April.

MR. PARLER: Off the record.

(Discussion off the record.)

MR. PARLER: Back on the record.

BY MR. PARLER:

Q In 1978 the General Accounting Office issued a report entitled "NUCLEAR POWERPLANT LICENSING: NEED FOR ADDITIONAL IMPROVEMENTS." The date of that report is April 27, 1978, and it has the identification "EMD-78-29."

Now, are you familiar with this report at all?

A I believe I read a draft of it. I am not sure I read the final. You know, these usually come to the agency in draft form.

Q Right. If you would go back to page 59, I believe that's the best place to start for what I want to do here -- and it's not going to be exhaustive.

Are you on page 59?

A Yes.

Q This is a letter which has in it the NRC's comments to the GAO, and it is a convenient reference where one can see both the pertinent recommendations as well as the agency's position on those recommendations.

At the bottom of the page there is a recommendation:

"That the Chairman, NRC: evaluate the scope and depth of reviews in the plant systems review branches to determine if additional staff or time are required to insure reviews are adequate."

Again, I am just using this as a point of departure. The question is: In view of your past experience in the systems review branches, is there a serious resource problem there, or what?

A There is. I think Dr. Mattson has taken steps to fix it. At the time this was written, the serious problem was that there was a gross undercalcuation of the length of time it took to review an operating license. The actual study, to my recollection, was done in the spring of '78 and showed we were actually spending between three and four times as much time to review an operating license as the model schedule allowed for. Somewhere in that time span we greatly expanded the amount of time we would give a review.

Q Would you turn over to page 65, Dr. Ross.

A Got it.

Q The recommendation there is:

"That the Cnairman, NRC: identify and meet the training needs of technical reviewers with special emphasis on (1) updating technical skills, (2) providing guidance on implementing the Standard Review Plan, and (3) providing an overall orientation of the licensing process and how each review section relates to an overall program to protect the public health and safety."

Do you have any comments on the state of affairs in that area, say around the first part of this year?

A Well, I never felt that our training skills program was deficient. I think there was adequate training done.

I believe on Item 2, that we give enough guidance on implementing the Standard Review Plan, we were very poor on giving ideas on how to be a regulator, which may be in there somewhere.

No. 3, for new people, is poorly done.

Q That is, for new people providing an overall orientation, and so on?

A Yes.

Q I only have a few others of these. On page 67 the recommendation is:

"That the Chairman, NRC: require technical reviewers to clearly document all conclusions, analyses, and review steps taken during the licensing review."

The question of documentation, I believe, if my recollection is correct, has been a longstanding one that people have been talking about.

Do you have any comments on that?

A I think that is a good recommendation. What it has to do with is discipline of an engineer or scientist. Some people in the review process already did that because they were well disciplined engineers. Some did it poorly because they weren't well disciplined.

I don't know why the Chairman should do it. It seems to me like an assistant director can do it, but I guess the buck stops with the Chairman.

Q Dr. Ross, in your deposition before the representatives of the President's Commission, in several places, one of which I believe is page 90, you referred to the fact that the staff's review of the Oconee operating reactor was where the detailed review of the particular B&W design that was involved in the Three Mile Island 2 accident too place, as well as a detailed review at the Three Mile Island operating license stage.

Is my characterization, in an attempt to summarize what you said, essentially correct as a point of departure here?

A Well, what I said was we did a lot of technical work on Oconee and didn't do it on plants that looked alike. And particularly there wwere a large number of topical reports written by B&W that approved the reactor. We read those and reviewed those, and it carried over into subsequent reviews.

Q You have also indicated that in various capacities you were either in a position to be involved or to be aware of, generally speaking, what were the significant things that were looked at in those reviews; is that right?

A That is correct.

Q Now, with this piece of background -- oh, there's another one.

Also in your testimony before the President's Commission representatives, in comparing the B&W reactor 177 design with others, you made the point, I believe, that it was sensitive or less forgiving to the disturbances than other plants; is that right?

A That is correct.

Q Now, with that background I can ask you the question, and realizing that it is a general question, that a lot of years have gone by.

But do you happen to recall whether the kinds of concerns that have been highlighted by the TMI accident with regard to the design of that plant were focused on to any great degree by any of the elements in the process? And by that I mean the staff review, the ACRS, Babcock & Wilcox?

A I don't think they were, no.

Q Dr. Ross, we will try in the time we have to cover some of the areas that appeared to us to be important for purposes of this special inquiry group and the recommendations about the regulatory program that the Director will eventually be called upon to make before the end of this year.

In that context, is there anything else that you would like to add that we have not successfully elicited for the record because of the questions that we did not ask, or the way that we asked the questions that we did? In other words, do you have anything else to add?

A About the only thing that comes to mind -- I said it somewhere else; I don't know whether it was in the other deposition or not; it doesn't matter -- is that the Commission and the reactor vendors turn out or publish a large amount of information annually on reactor safety and research. That information is probably not disseminated as widely or evaluated as fully as it ought to be in order for us to better do our job.

I will just lay the problem out. There are several solutions.

I think the first step is to recognize that it is a problem and just let it go at that.

I think that is the only thing I would add.

Q Thank you.

Dr. Ross, in conclusion let me say that this is an ongoing investigation, and although we have completed the questions we have for you today, we may need to bring you back for further depositions. We will, however, make every effort to avoid having to do so.

I will now recess this deposition, rather than terminate it. We wish to thank you for your time in being here today, and for your cooperation and contributions.

Thank you, sir.

A I have one more thing I would like to have on the record.

Q All right.

A I have four IOUs, I mean documents that I have got to go back and dig up and send. To whom do I send them?

Q Mr. Richard DeYoung.

A Also I would like to say on the record that I have the office he used to have, and the roof still leaks. I want that on the record.

(Laughter.)

Q I can add for the record that Mr. DeYoung's temporary office still leaks, too.

A Okay.

MR. PARLER: Thank you, sir.

(Whereupon, at 12:50 p.m., the deposition was recessed.)

** ** **

Exhibit 1151

September 6, 1979

In Reply Refer to:
NTFTM 790906-03

Dr. Denwood Ross, Deputy Director
Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

POOR ORIGINAL

Dear Dr. Ross:

I am writing to confirm that your deposition under oath in connection with the accident at Three Mile Island is scheduled for September 28, 1979 at 9:00 a.m., in Room 6715, Maryland National Bank Building. This will also confirm my request for you to have your resume and any documents in your possession or control regarding TMI-2, the accident or precursor events which you have reason to believe may not be in official NRC files, including any diary or personal working file.

The deposition will be conducted by members of the NRC's Special Inquiry Group on Three Mile Island. This Group is being directed independently of the NRC by the law firm of Rogovin, Stern and Huge. It includes both NRC personnel who have been detailed to the Special Inquiry Staff, and outside staff and attorneys. Through a delegation of authority from the NRC under Section 161(c) of the Atomic Energy Act of 1954, as amended, the Special Inquiry Group has a broad mandate to inquire into the causes of the accident at Three Mile Island, to identify major problem areas and to make recommendations for change. At the conclusion of its investigation, the Group will issue a detailed public report setting forth its findings and recommendations.

Unless you have been served with a subpoena, your participation in the deposition is voluntary and there will be no effect on you if you decline to answer some or all of the questions asked you. However, the Special Inquiry has been given the power to subpoena witnesses to appear and testify under oath, or to appear and produce documents, or both, at any designated place. Any person deposed may have an attorney present or any other person he wishes accompany him at the deposition as his representative. The Office of the General Counsel of NRC has advised us that it is willing to send an NRC attorney to all depositions of NRC employees who will represent you as an individual rather than represent NRC. Since the NRC attorney may attend only at your affirmative request, you should notify Richard Mallory (634-3224) in the Office of the General Counsel as soon as practicable if you wish to have an NRC attorney present.

You should realize that while we will try to respect any requests for confidentiality in connection with the publication of our report, we can make no guarantees. Names of witnesses and the information they provide may eventually become public, inasmuch as the entire record of the Special Inquiry Group's investigation will be made available to the NRC for whatever uses it may deem

OFFICE ▸						
SURNAME ▸						
DATE ▸						

appropriate. In time, this information may be made available to the public voluntarily, or become available to the public through the Freedom of Information Act. Moreover, other departments and agencies of government may request access to this information pursuant to the Privacy Act of 1974. The information may also be made available in whole or in part to committees or subcommittees of the U.S. Congress.

If you have testified previously with respect to the Three Mile Island accident, it would be useful if you could review any transcripts of your previous statement(s) prior to the deposition.

Thank you for your cooperation.

Sincerely,

/s/

Mitchell Rogovin, Director
NRC/TMI Special Inquiry Group

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POOR ORIGINAL

OFFICE	NRC/TMI	NRC/TMI	NRC/TMI	NRC/TMI	NRC/TMI	NRC/TMI
SURNAME	WParler:kr:mc	PNorry	RDeYoung	EKCornell	GFrampton	MRogovin
DATE	9/ /79	9/ /79	9/5/79	9/5/79	9/ /79	9/ /79

Exhibit
1152
9/28/79

PERSONAL QUALIFICATIONS
DENWOOD F. ROSS, JR.

I am presently employed with the U.S. Nuclear Regulatory Commission as the Deputy Director for the Division of Project Management, Office of Nuclear Reactor Regulation. However, I have been detailed for 6-8 months service as Director of the Bulletins and Orders Task Force in NRR. My work address is 7920 Norfolk Avenue, Bethesda, MD.

My previous job assignment, as Assistant Director for Reactor Safety (from 1/76 to 10/78), included supervising the activities of the Analysis Branch, the Core Performance Branch, and the Reactor Systems Branch which, together, form the Reactor Safety group in DSS. The work assignments performed by Reactor Safety included evaluation of emergency core cooling system response, as well as reactor core and primary coolant system response to transient and other accident conditions.

Prior to that assignment I served as the branch chief of the Core Performance Branch for about 2½ years. Other job assignments since coming to USNRC (then AEC) in August 1967 include project manager assignments for several projects, including Three Mile Island (Units 1 & 2), Crystal River Unit 3, Oconee 1, 2, and 3, and Quad Cities 1 & 2. In addition, I served on a special task force reviewing ECCS performance, including extended service at the ECCS rule-making hearing.

Prior to joining NRC I worked at the General Dynamics nuclear research facility at Ft. Worth, Texas for 10 years, including 4 years as operating supervisor for three research and test reactors. I also worked for 1½ years at the NTR-ETR operations at the NRTS, Idaho.

I have degrees in Civil Engineering (BS, 1953), Mathematics (MS, 1963), and Nuclear Engineering (MS, 1960, and D. Engr., 1974).

Exhibit 1153

Westinghouse Nuclear Europe

Castle, W J Esposito info

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H. Cordle (with att.)
D. ten Wolde (with att.)
A. Hall (with att.)
T. Currie (with att.)
J.P. Lafaille (with att.)
R. Galletly (with att.)
R. Lehr (with att.) Pitts.
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From : T. Cecchi
Date : September 4, 1974
Ref : SA/251

POOR ORIGINAL

SUBJECT : TECHNICAL REPORT ON NOK 1 INCIDENT OF AUGUST 20, 1974

References (1) Telex SE-G-74-195 (8/26/74) to NOK by H. Cordle
(2) Letter (8/27/74) NKA-3940 from L. Barshaw.

You will find attached the technical report on NOK 1 Incident of August 20, 1974 prepared by WNE inspection team who went to Beznau on August 23.

This report, which should be sent to Beznau, summarizes our observations on the course of the transient, the damage as we viewed it, our calculations and conclusions.

Despite what is indicated in the referenced (2) letter, in order to have a more complete report, we added some recommendations for future changes.

T. CECCHI
SYSTEMS ANALYSIS

SA/251

TECHNICAL REPORT ON BEZNAU UNIT ONE

INCIDENT OF AUGUST 20, 1974 : TG-1 TRIP/

REACTOR TRIP/SAFETY INJECTION ACTUATION.

J.P. LAFAILLE
R. GALLETLY
T. CECCHI

H. CORDLE, Director,
Systems Engineering

September 2, 1974

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TECHNICAL REPORT ON BEZNAU UNIT ONE INCIDENT OF AUGUST 20, 1974 : TG-1 TRIP/REACTOR TRIP/SAFETY INJECTION ACTUATION

I - INTRODUCTION

This report is produced as a result of a site visit following the incident on Beznau I which took place on August 20, 1974. The object of the visit was to make a rapid evaluation of whether the consequences of the incident would jeopardize safety. This report confirms the telex of Aug. 28, 74 on this subject.

The scope of this report, therefore, is limited to a description of the sequence of events and of the damage observed together with a possible explanation and assessment of safety issues. It is not meant to be a comprehensive analysis of the effects of the incident.

II - SEQUENCE OF EVENTS DURING THE INCIDENT

On August 20, 1974, a trip of one of the two turbines on the Beznau I reactor followed by failure of the steam dump system to operate resulted in a reactor trip and the opening of the pressurizer relief valves. One of these valves subsequently failed to close and the extended blowdown of the pressurizer resulted in the rupture of the pressurizer relief tank, bursting disk. Examination following the incident revealed that the pressurizer relief valve which had failed to close had been damaged, as had some of the supports to the pressurizer relief line itself.

The sequence of events, with times where known, is reconstructed below :

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Initial conditions :

Date : August 20, 1974 Time : 11.20 a.m.
Pressurizer pressure : 154 bar Pressurizer level : 50%
Pressurizer relief tank level : 80%
Power output of turbogenerator 1 : 187 MW (e)
" " " " 2 : 177 MW (e)

Time	Event
	Disturbance occurs on the external grid network.
	TG1 trips out on high casing vibration.
11 hrs 20 min 07.8 sec	Vibration causes low Δ p signal from hydrogen seal oil system.
	Steam dump valves fail to open.
	SG steam pressures rise.
	Pressurizer pressure rises.
	Pressurizer level rises.
20 11.9	Both pressurizer relief valves open.
20 17.8	Turbotrol of TG2 drops into the emergency mode.
20 23.0	One pressurizer relief valve closes in accordance with automatic signal, pressure continues to fall and level continues to rise.
	Pressurizer relief tank pressure rises.
	Pressurizer relief tank level rises.
	TG2 power level falls then rises to an overpower of 214 MW (e).
21 00.4	Reactor trips on pressurizer low pressure.
21 01.2	TG2 trips.
	SG steam pressures rise.
	SG water levels fall.
	Pressurizer level falls.
23 03.5	Secondary side safety valves lift.
23 13.9	Steam is formed in the RCS hot legs and pressurizer level rises past 100% and remains off-scale for 3 to 5 minutes.
	A reasonable assumption is that water discharge occurs through the open relief valve.
	Operator shuts pressurizer relief line isolation valve. (Reported verbally as 2 to 3 minutes after the trip).

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	Pressurizer level falls rapidly as steam bubbles in RCS collapse.
	Pressurizer relief tank bursting disk ruptures.
	Pressurizer relief tank pressure falls.
	Pressurizer relief tank level falls.
11 hrs 23 min 43.5 sec	High containment pressure recorded (peak 1.1 bar abs.).
24 51.2	High containment temperature recorded (53.4°C).
25 17.8	High containment activity recorded (17.3 mr/hr).
32 14.3	SIS initiated as pressurizer level falls to 5%.
	Pressurizer level rises as SI water is added to the RCS.
	SIS stopped manually.
Subsequently	Procedure begun to bring reactor to cold shutdown condition, using the atmospheric steam relief valves.

Fig. 18 shows the record of pressurizer pressure and level transients following incident initiation.

III - TRANSIENT BEHAVIOR OF MAIN PLANT VARIABLES DURING THE INCIDENT

A turbine trip in a two turbine plant is equivalent to a 50% load rejection and no reactor trip should be initiated if control systems work correctly. Since in Beznau I the steam dump system did not work at all, initially the main variables behaved as follows :

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1. Steam Generator steam pressure rose (to about 66 bars) but not enough in order to actuate safety valves.

2. Feedwater flow, steam flow and steam generator level decreased normally as expected.

3. The reactor being in automatic control, the nuclear power decreased. When reactor was tripped after about 49 seconds, it was at 76%.

4. Pressurizer pressure rose rapidly from 154 bars to a maximum of 160 bars (pressurizer relief valves actuation) in about 11 seconds.

5. Reactor coolant system average temperature rose rapidly from 298.5°C to a maximum of 305.5°C in about 50 seconds.

6. Cold leg temperature rose rapidly from 275°C to 290°C, then decreased to 240°C in 10 minutes, to 220°C in next 100 minute and to 140°C in next 170 minutes.

7. Pressurizer level rose from 50% to 67% in about 50 seconds.

Due to the fast pressurizer pressure increase, both pressurizer relief valves were rapidly actuated. Their actuation took place almost simultaneously. However, it is very probable that the valve actuated by the compensated pressure error signal (signal elaborated by a PID controller) opened some seconds before the other one due to the derivative term of the PID controller.

When pressure decreased below relief valves actuation setpoint the valve directly controlled from an uncompensated pressure signal did not shut. This resulted in a depressurization at rate of about 0.75 bar/sec, resulting in a reactor trip by low pressu[re] in approximately 49 seconds.

The reactor trip signal tripped the turbine which was still in operation, resulting in a further steam pressure increase (above 70 bars) which produced steam generator safety valves actuation, lowering the pressure to about 65 bars.

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Reactor coolant system average temperature decreased to about 285°C and pressurizer level to 23% in about 1 minute after reactor trip. At this point pressurizer pressure had fallen to hot leg saturation (70 bars). Subsequently, hot leg flashing resulted in an increase of pressurizer level until the pressurizer filled about 3 minutes after reactor trip, resulting in probable liquid water discharge from the relief valve and bulk boiling in the core. x Then the operator isolated the failed relief valve, and pressurizer level decreased reaching the setpoint (5%) for safety injection actuation (safety injection is actuated by coincident low pressurizer pressure and level S.I. signals) about 11 minutes after reactor trip. The system then started refilling. When pressurizer level reached about 70%, safety injection pumps were shut off manually.

The reactor was then brought normally to cold shutdown conditions.

IV - DAMAGE TO THE RELIEF PIPE RESTRAINTS AND SUPPORTS

For pipe layout, see isometric, fig. 1 attached.

The relief line to the power relief valves comes out of the pressurizer top and runs directly down (vertical run of 6.8 m). It passes through a grating floor. No impact evidence between the floor and the pipe insulation exists. (Gap about 25 mm). At the bottom of the vertical run there is a console type restraint. (Location 1 in fig. 1). The main dimensions are given in fig. 2. There is contact evidence, as shown on the figure, but no damage.

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The pipe then runs horizontally to the restraint 2 (fig. 1). This restraint limits motion of the pipe in a horizontal direction, perpendicular to the pipe axis (See fig. 3). Scratches on the shoes indicate that the pipe moved about 26 mm axially. The top part of the insulation is slightly smashed (See fig. 3).

../...

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../...

x Nuclear Power was then at [illegible]%.

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The line then runs vertically down (2.77 m) and separates into two branches each having a stop valve and a relief valve. Fig. 7, 8 and 9 show the damage to the valve.

Examination of the pressurizer relief valve which failed to close revealed that the yoke had broken off completely. One arm of the cast iron yoke had broken at the top and the other arm at the bottom taking part of the yoke ring with it. The top break showed the presence of a very large flaw (inclusion). All broken faces showed classic brittle failure together with evidence that the faces had rubbed together following failure. In addition it was reported that the valve spindle had been slightly bent. This was not observed since repairs had already been started.

Fig. 6 and 7 show the pedestal of the support between the two valves. Fig. 4 is a sketch of the support and details the damage.

The damage corresponds to a rotation of the pipe around a horizontal axis perpendicular to the pipe axis. No evidence of translation has been found. Considering fig. 7, the back bolts were strained much more than the front ones.

The bolts of the undamaged valve support have been inspected. It was found that the paint was cracked at the bolt joints, but no other damage could be found.

After the valves the two branches of the pipe drop to the lower floor. Fig. 10 shows the penetration corresponding to the damaged branch.

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At the lower floor, the restraint R4 (See fig. 1) has been pulled off the floor (see detail in fig. 14). The motion has been imposed on the frame by the bar of the hanger passing through a 50 mm slot in the frame (See fig. 11).

Restraint R5, which is only a column supporting a sliding shoe, shows a motion of 70 mm as shown in fig. 5.

The pipe then joins a header and passes through the floor (R6 on fig. 1). There is evidence of 25 mm upward displacement.

At the lower floor the header has an elbow. Motion is restrained by a snubber. The bolts fixing the snubber to the concrete were found to be loose.

V - EVALUATION OF THE INCIDENT

This evaluation covers the incident transient effects and a preliminary estimate of magnitude and probable causes of damage to the pressurizer relief piping and supports.

1. Comparison with design transients

This Beznau I incident is similar to the two following incidents which are normally considered among reactor coolant system design transients :

- Loss of load (up to pressurizer relief valves actuation).
- RCS depressurization (from pressurizer relief valves actuation).

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From the standpoints of core power, heat transfers and systems pressures and temperatures, the reported incident is less severe than the design transients considered above.

The magnitude and variation rate of the temperature and pressure transients resulting from the incident are indeed fully covered by the values used for equipment design.

Plant variable behavior during the transient did not result in an uncontrolled or damaging situation, and the released activity remained well below dangerous limits. All existing protection systems (steam generator safety valves, reactor trip, safety injection) worked properly and were adequate to handle the incident avoiding core and equipment damage.

2. Evaluation of damage to the pressurizer relief line, the relief valves and supports.

The relief line between the pressurizer and the power relief valves is part of the reactor coolant pressure boundary and therefore is important to the safety of the plant.

The one power relief valve which failed to close was isolated in accord with design intent by the operator closing the appropriate relief isolation valve and hence no uncontrolled loss of coolant occurred.

The review of the relief line equipment showed damage to the relief line supports and the pressurizer relief valve PCV-456.

The damage evaluation and probable causes are treated below.

a) Discussion of the incident related to cause of damage.

Examination of the relief line and supports along with the records of primary reactor coolant system parameters leads to the following observations.

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(1) It is probable that the observed damage to the supports is the result of hydraulic shocks from a sequence of water and steam discharge through the relief line.

(a) The pressurizer relief line from the relief valve to the pressurizer can fill with condensate. This distance is approximately 19 meters, and can contain a volume of 0.06 m^3. Opening of the relief valves will cause a rapid discharge of the water. The resulting dynamics are one possible cause of the piping displacements observed.

(b) Based upon the recorder chart of pressurizer water level, it appears probable that some water discharg occurred later in the transient when the pressurize was completely filled. The records indicate that this event could only have occurred after automatic closure of the undamaged valve (PCV-455C).

Dynamics related to this event are another possible cause of the observed piping displacements and support damage.

(2) It is not possible from available evidence to provide one sequence of events which uniquely explains the observed results of the transient.

It is not certain that the valve damage was the consequence of the same hydraulic shock that resulted in the support damage.

The observed sequence of events indicates that one likely scenario is as follows :

(a) The undamaged relief valve, PCV-455C, opens first on the derivative compensated pressure controller a few seconds before the second valve opens.

(b) The water slug formed by condensed pressurizer steam in the relief line is largely discharged through the undamaged valve. We note that this portion of the line showed little or no support damage.

(c) The second valve, PCV-456, opens on continued pressure increase and the transient, combined with the large flaw in the valve yoke results in valve failure.

With this hypothesis, there is no reason to expect a hydraulic shock higher than in opening of the first valve hence piping displacement sufficient to damage supports might not yet have occurred.

(d) The first valve closes automatically upon a reducing pressure signal before pressurizer water level reaches 100%.

(e) Water discharge occurs upon filling the pressurizer creating a substantial hydraulic shock in the relief line. Since the undamaged valve has already closed, the resultant pipe displacement was most pronounced in the portion of line where the damaged valve is located.

Other scenarios can also be postulated, but none has sufficient support of evidence to permit identification of a single sequence of events as the cause of observed damage.

(3) The events which lead to complete filling of the pressurizer and the second water discharge through the relief line required more than a single failure :

(a) The failure of all the secondary steam dump valves to operate.

(b) The failure of the pressurizer relief valve to close. It is likely that such a failure would not have occurred even with an initial hydraulic shock without existence of a large flaw in the relief valve yoke.

(4) Considering the valve PCV-456 itself, when in the open position, there is a spring force producing a tension of about 60,000 to 80,000 Newtons in the yoke. When the disk lifts, this force can be amplified due to dynamic effects. The presence of the flaw in one of the arms overstressed that arm (area reduction and stress concentration), which caused it to break.

This caused a moment to be applied to the other arm, resulting in bending of the spindle and rupture of the base. The broken metal surface appearance was typical of brittle failure with some polishing due to rubbing contacts following yoke separation. The yoke the rose about 2,5 cm, the normal stroke of the valve. With the broken yoke, the valve failed to close. Dynamic forces due to the free motion of the operator body may have contributed to damage to the support.

(5) Appendix A calculates the forces and stresses on the relief line piping in two locations, suspected to be among the most stressed. It is seen there that, within the calculation assumption the piping could have been marginally overstressed. However, since a dye penetrant check of the PVC-456 valve to pipe weld was reported to show no defect, we cannot see any reason to think that the plant would operate in unsafe condition with the line in the present state. This statement assumes of course that all the support system of the piping will have been returned to its design condition before the reactor goes back to power.

To gain further assurance on the safety of the line we would recommend that a dye penetrant check of all welds near the fixed points be made at the earliest convenience. The locations include the pressurizer nozzle, the relief tank nozzle and the intermediate supported or restrained points.

b) Operational Considerations

(1) Plant operation with one pressurizer power relief valve closed off does not present a safety problem. The high pressure reactor trip and the pressurizer safety valves provide the necessary protection against overpressure of the reactor coolant pressure boundary.

The existence of the power relief valves is to prevent unnecessary opening of the main code safety valves during certain plant design transients.

(2) The safety injection system functioned normally with a reported total injection rate of 40 l/sec. The injected water raised the pressurizer level from 5% to 75%. Assuming the injection water to be initially at 16°C and atmospheric pressure in the RWST and to end up in the pressurizer at 285°C and 110 bars then the quantity of water leaving the RWST must have been about 10 m^3. This would cause a decrease in RWST level of about 0.7%. The injection time would be about 4.1/2 minutes assuming a constant injection rate.

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(3) The reason why the turbotrol gear of turbine 2 dropped into the emergency mode is not known. It was reported that the effect of this would be to lock the turbine inlet control valves in their last position. Thus they would no longer respond to changes in steam pressure. This may account for the overpower excursion experienced on turbogenerator 2 just prior to its tripping.

(4) The failure of the steam dump valves to open was reported to be the result of a wrong wiring connection which was not discovered during testing. The control circuitry of the steam dump valves had been out for maintenance at some previous date. Before being put back on line, the circuitry had been tested in two halves. Each half was checked independently of the other half and each half checked out satisfactorily. A fault at the interface of the two halves thus remained unrevealed.

VI - OTHER RECOMMENDATIONS

1. The piping displacements and support damage which occurred have indicated the possibility that the pressurizer relief line was marginally overstressed. The likelihood is that the displacements resulted from either discharge of a water slug initially in the line or from relief of water when the pressurizer was completely filled.

   The initial evaluation of stress was deduced from observed support displacement and support bolt strains. As such, no definitive indication of possible stress levels with this transient exists as basis for an evaluation of fatigue damage for the entire piping length.

   We would recommend a dynamic analysis be performed, considerir at a minimum the effects of the steam condensate initially in the line. The force time history function can then be used for luation of fatigue damage as well as the adequacy of restraints.

2. The failure of the power relief valve yoke is more probable due to the use of cast-iron materials of construction where impact properties are poor and flaws of the type involved in this failure can remain undiscovered.

   We therefore recommend such non-destructive tests as are feasible be made to ascertain that no flaws of this type exist in the valve currently installed.

   Further consideration might be given to replacing these yokes with a less brittle material.

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3. The test procedures following maintenance of the control system to the steam dump valves should be rewritten to eliminate the possibility of unrevealed faults.

4. It would be useful to provide means (i.e. 2 separate alarms : one actuated by the uncompensated pressure signal and the other by the compensated pressure error signal) in order to know if certainly each pressurizer relief valve opens during a pressure excursion.

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APPENDIX A

Stress and Force Evaluation in the pipe between valves

1. Damage to the support

The two bolts on the right side on figure 3 were strained about 3 mm. The two bolts on the left side were also strained but only to the point of getting loose.

2. Evaluation of the moment applied to the support

Bolt size : M10 → Shaft size (diameter)

$$8.888 < d < 9.128 \text{ mm}$$

(Catalogue MARC-GERARD - 1970)

Section (average) $\frac{\pi}{4}\left(\frac{8.888 + 9.128}{2}\right)^2 = 63.73 \text{ mm}^2$

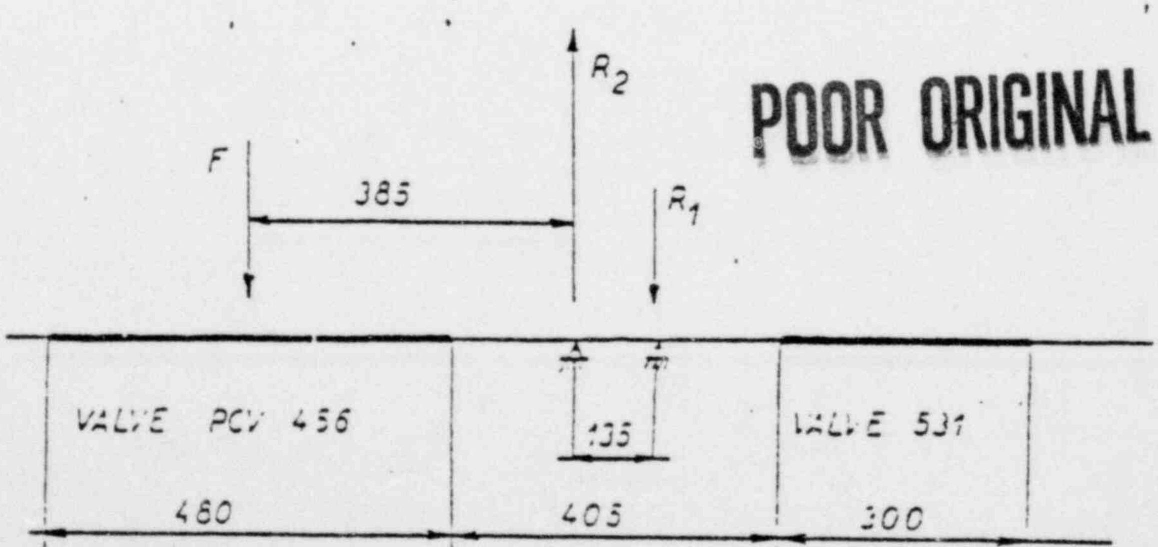
Assume for the bolt material a yield stress of

$$\sigma_Y = 32 \text{ kg/mm}^2$$

Hence the moment to strain the two bolts is

$$M = 63.73 \times 32 \times 2 \times .135 = 550.6 \text{ kg.m}$$

3. Force required to create that moment



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If one neglects the effect of the supports located downstream of valve 456, one can write the equation

$$385*F = 135*R_1$$

Knowing that $R_1 * .135 = 550.6$ kgm

Hence $F = 1430$ kg

It is felt that such a force is in the possible range.

4. Stresses in the pipe (Primary stresses only)

Pipe : 3" sch 160

Hence : OD = 3.5 in = 88.9 mm   t = 11.13 mm

$$\text{Bending modulus} = \frac{I}{v} = 47.17\ 10^3\ \text{mm}^3$$

Bending stress :

$$\sigma_B = \frac{M}{I/v} = \frac{550.6\ 10^3}{47.17\ 10^3} = 11.67\ \text{kg/mm}^2$$

Pressure stress (ASME III, Article NB 36 52)

$$\sigma_P = \frac{p*OD}{2t} = \frac{164.5*10^{-2}*88.9}{2*11.13} = 6.57\ \text{kg/mm}$$

Combination (Article NB 36 52)

$$B_1 \frac{PD_o}{2t} + B_2 \frac{D_o}{2I} M_i$$

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B_1 and B_2 are taken from table 3683.2-1

$B_1 = B_2 = 1$

Hence $\sigma_{tot} = 6.57 + 11.67 = 18.24\ \text{kg/mm}^2$

5. Allowable stresses

SA 376 Grade 316

S_m at room temp. = 20 ksi = 14 kg/mm^2

S_m at 650°F (=343°C) = 16.6 ksi = 11.6 kg/mm

Allowable stress = 1.5 S_m (ASME III, article NB 36 52)

1.5 S_m = 21 kg/mm^2 (room temperature)
= 17.4 kg/mm^2 (343°C)

6. Conclusion for primary stresses in the pipe

Since it appears that hot fluid has been carried by the pipe for a time of about 3 min, the hot allowable stress needs to be taken. Then it appears that the actual stress is slightly higher than the allowable :

$$18.24 > 17.4 \; kg/mm^2$$

It should be noted that the figure of 18.24 kg/mm^2 is a minimum, since it corresponds to the plastification of the support (M = 550.6 kgm).

7. Primary and Secondary stresses in the pipe

The evaluation of secondary stresses (article NB 3653.1) requires the knowledge of the temperature gradients in the pipe. It was thus not possible to evaluate these stresses.

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8. Primary stresses at the reducer

Bending moment

$$M = 1430 \times \left(385 - \frac{1}{2}(405 - 135)\right) \; kg\,mm$$

$$= 357 \; kgm$$

reducer $2\frac{1}{2}$ " sch 160

OD = 2.875 in = 73.02 mm t = .375 in = 9.52 mm

$\frac{I}{v}$ = 1.64 in^3 = 26.9 cm^3

Pressure stress = $\frac{p \times OD}{2t}$ = 6.28 kg/mm^2

Bending stress = $\frac{M}{I/v}$ = 13.28 kg/mm^2

Total stress = 19.56 kg/mm^2

This stress should be considered more as indicative since it depends so much on the assumption of the force location.

The same conclusion holds as for the pipe stress.

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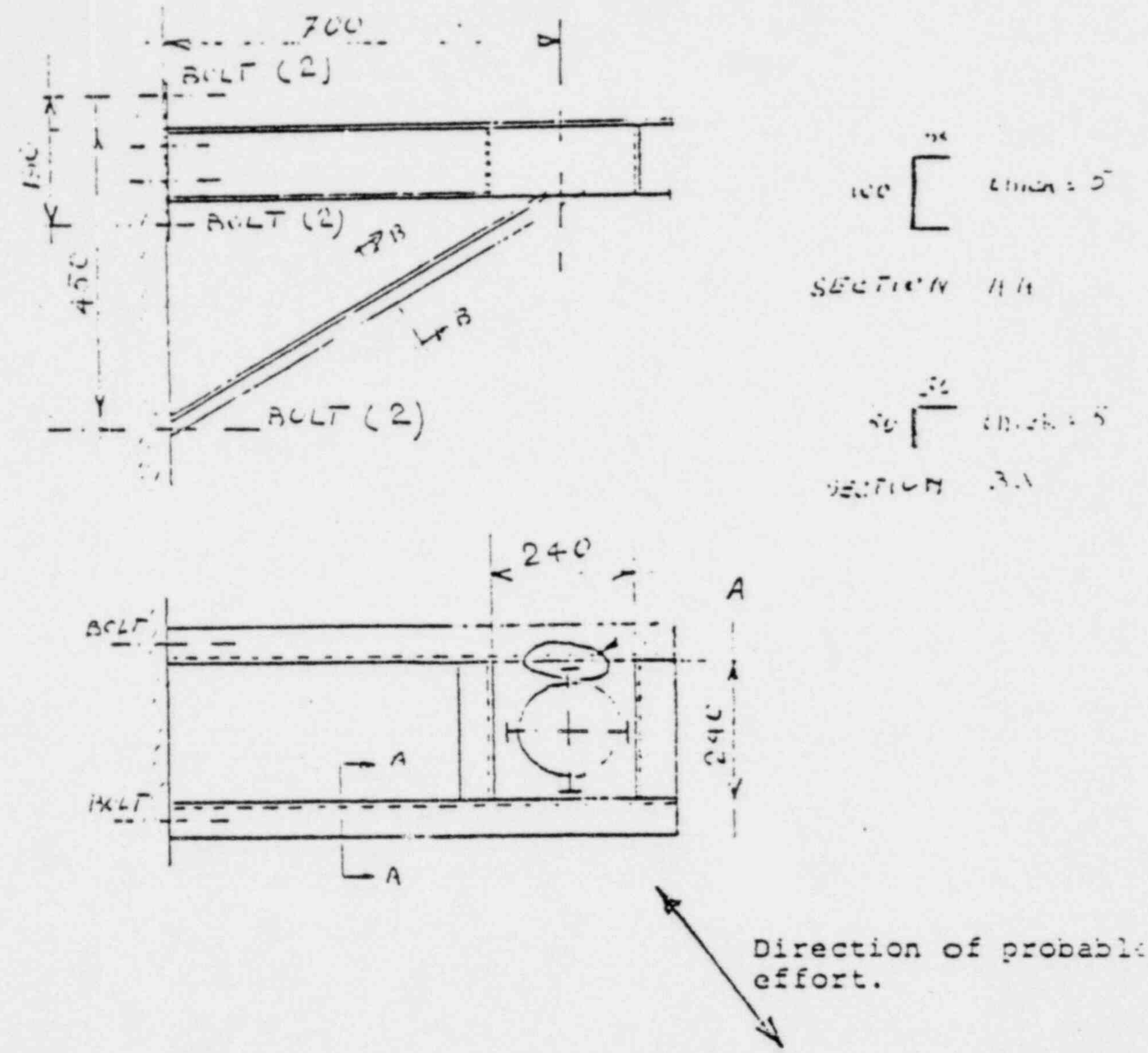

Direction of probable effort.

Bolts (6 total) : Hexagonal head = 25 mm

Damage : - no general distortion
- no rubbing evidence
- contact evidence in A

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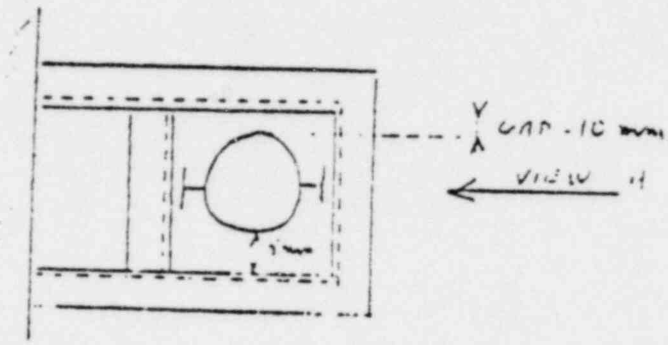
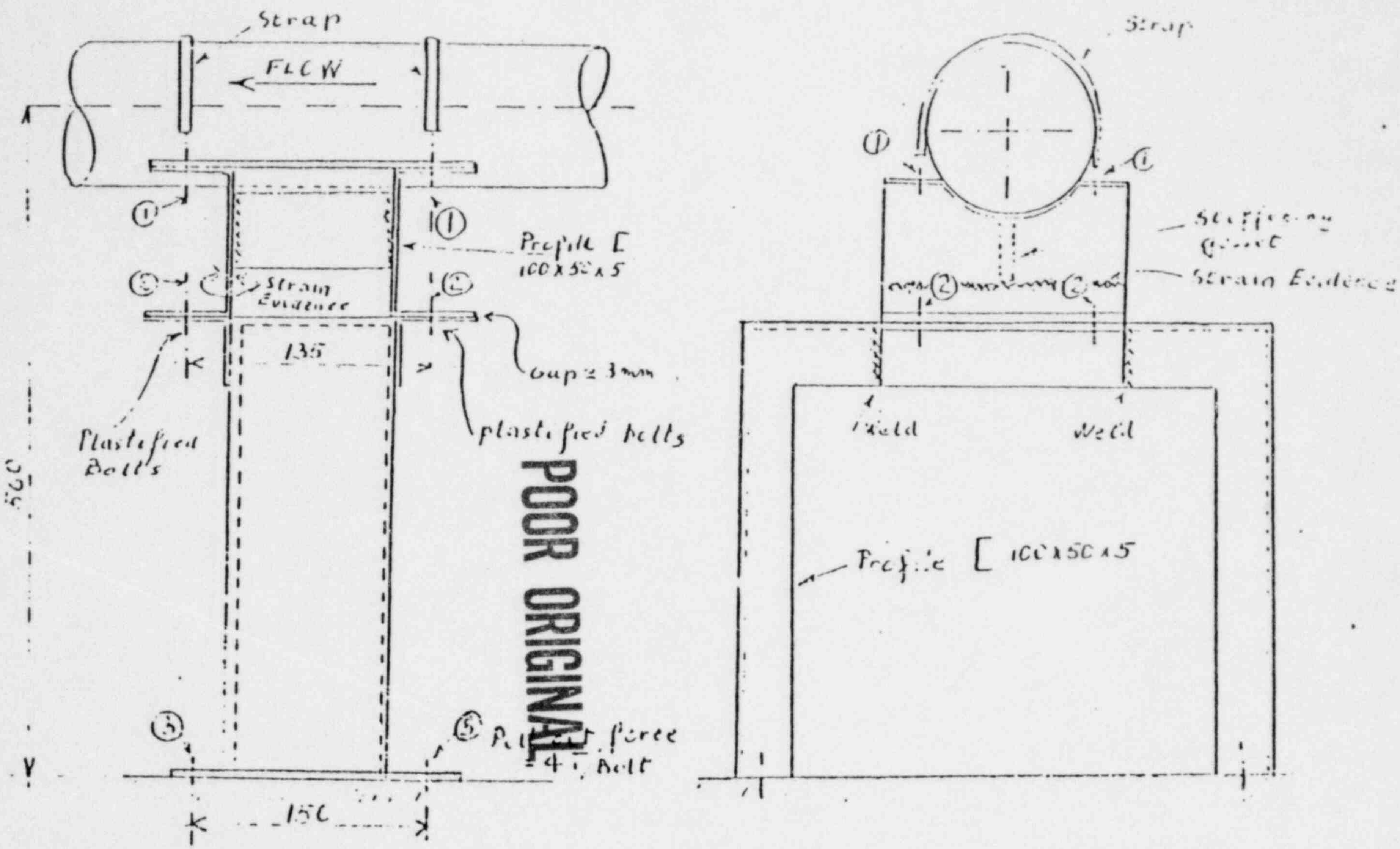
Figure 2 - Restraint R-1



View A (marks on the shoes)

Damage : - top of insulation slightly smashed
- scratches on shoes as shown on view A

Figure 3 - Restraint R-2



Bolts 1 (4 total) M-10
2 (4 total) M-10
3 (4 total) pull out

force = 4T/bolt

Damage : - no evidence at straps, pipe and bolts (1) and (3)
- all 4 bolts (2) have been strained
- gap measured as shown
- strain evidence in the [profile as shown

Figure 4 - Restraint R-3

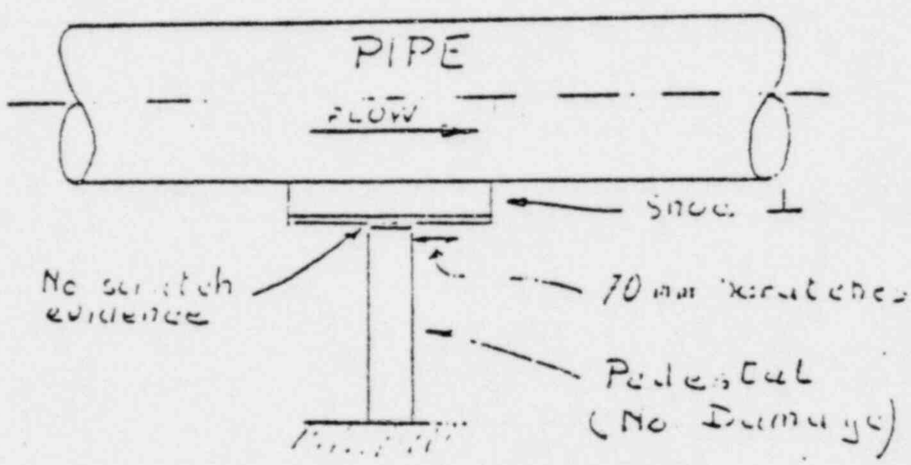


Figure 5 - Restraint R-5 Motion Evidence

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

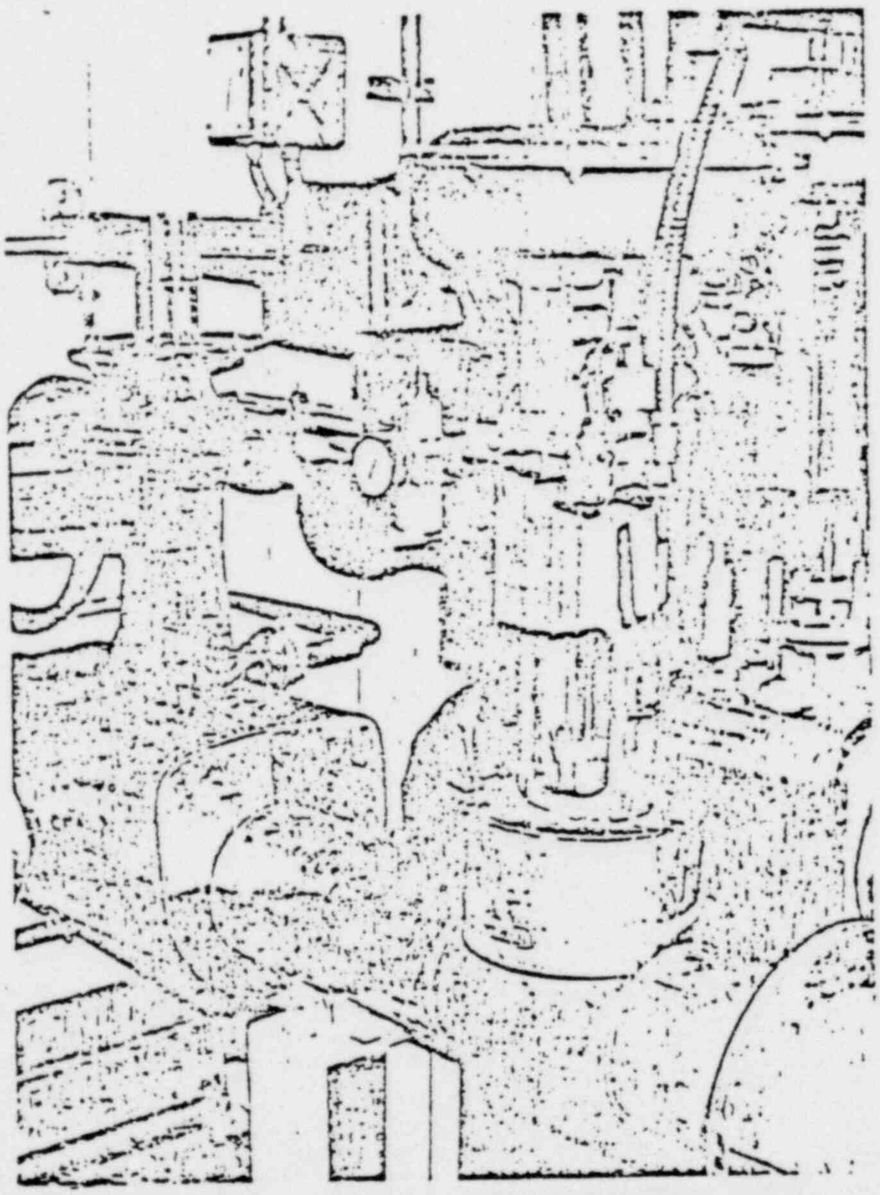


Figure 6 - Undamaged Relief Valve.

B_ [illegible] (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

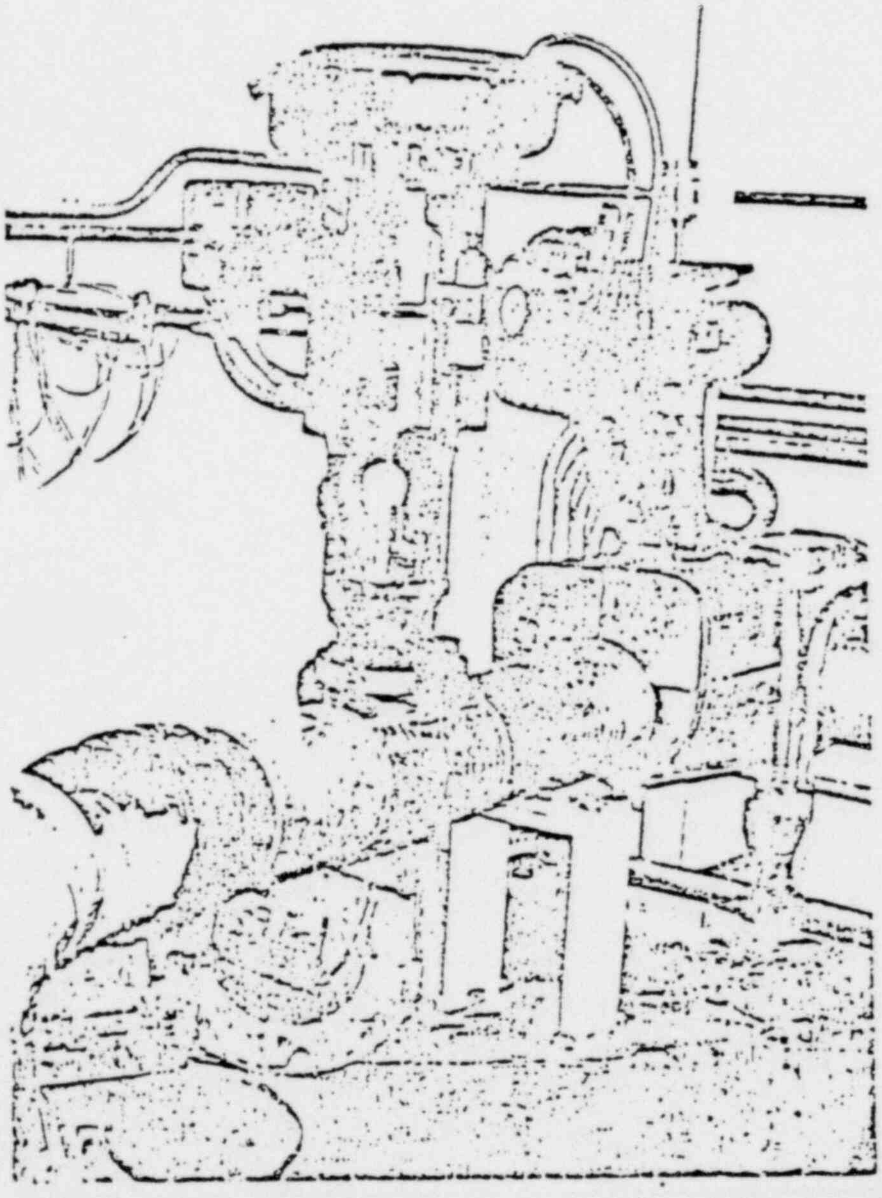


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Figure 7 Damaged relief valve
General view showing the two fractured arms
and the liefted operator.

BEZNA. - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

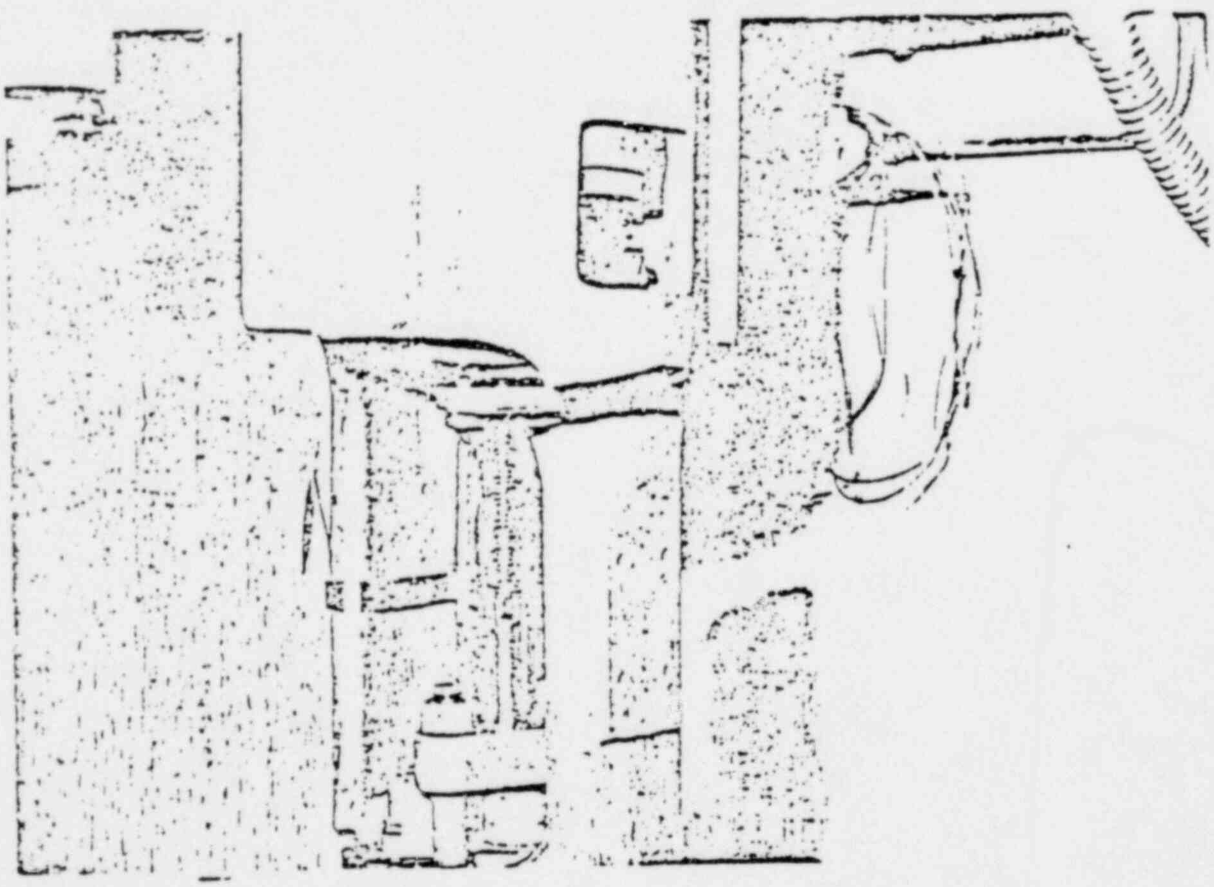

Figure 8 - Damaged Valve.

Detail of fractured yoke

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BEZNAU - UNIT N° 1 . (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

Figure 9 - Damaged Valve.

Detail of fractured bonnet.

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE.

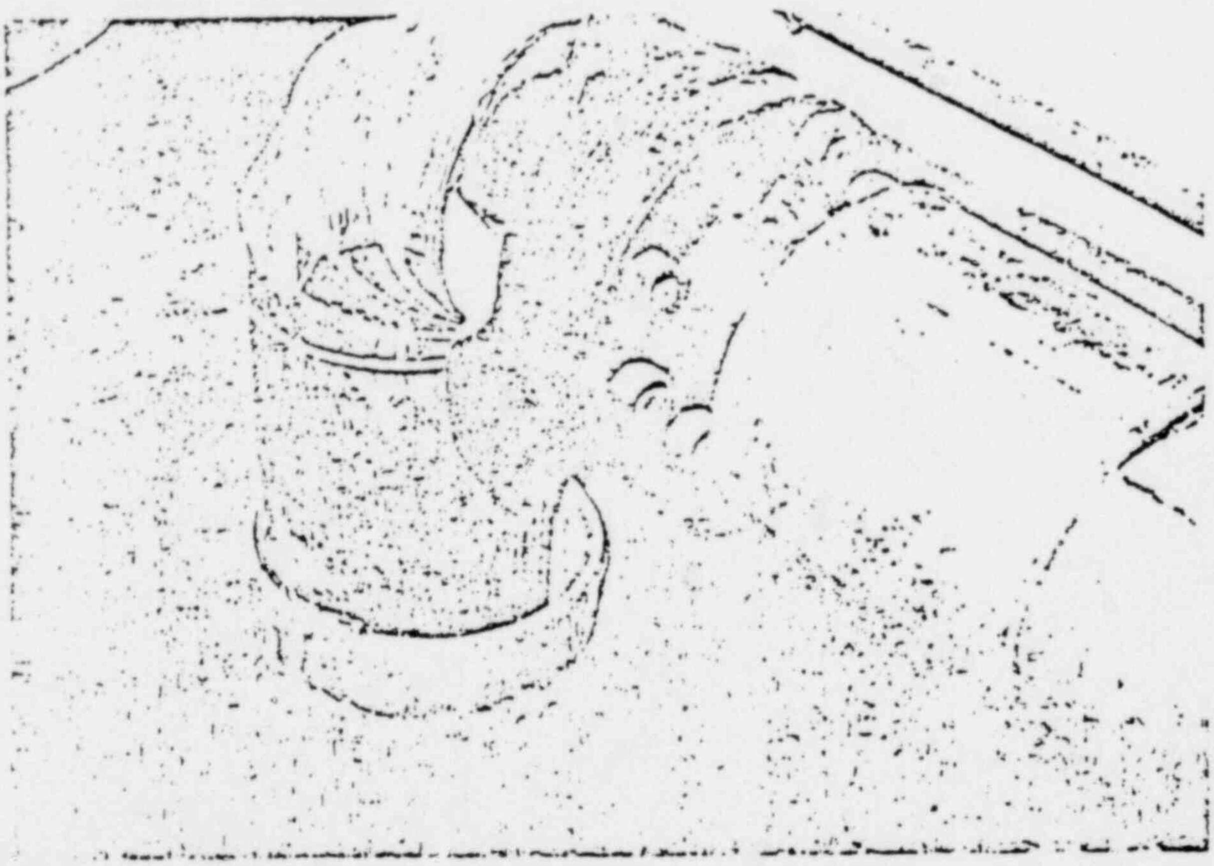
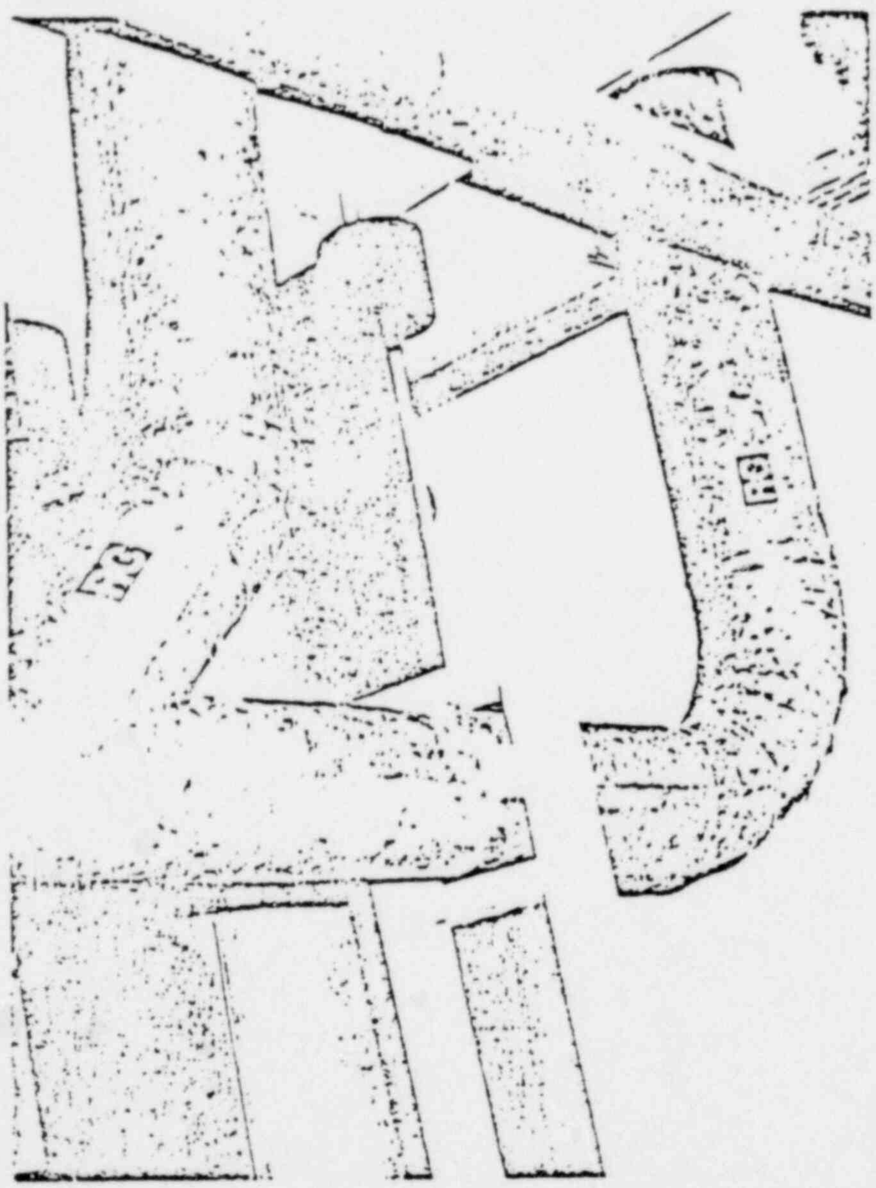


Figure 10 - Elbow after damaged valve.

POOR ORIGINAL

BEZNAU - UNIT N° 1 (NO.
STEAM DUMP FAILURE INCIDENT
Aug. 21, 74
PRESSURIZER RELIEF LINE



POOR ORIGINAL

Figure 11 - Support R4 (1)
General arrangement

100 x 50 x 5 profiles
50 mm slot

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE

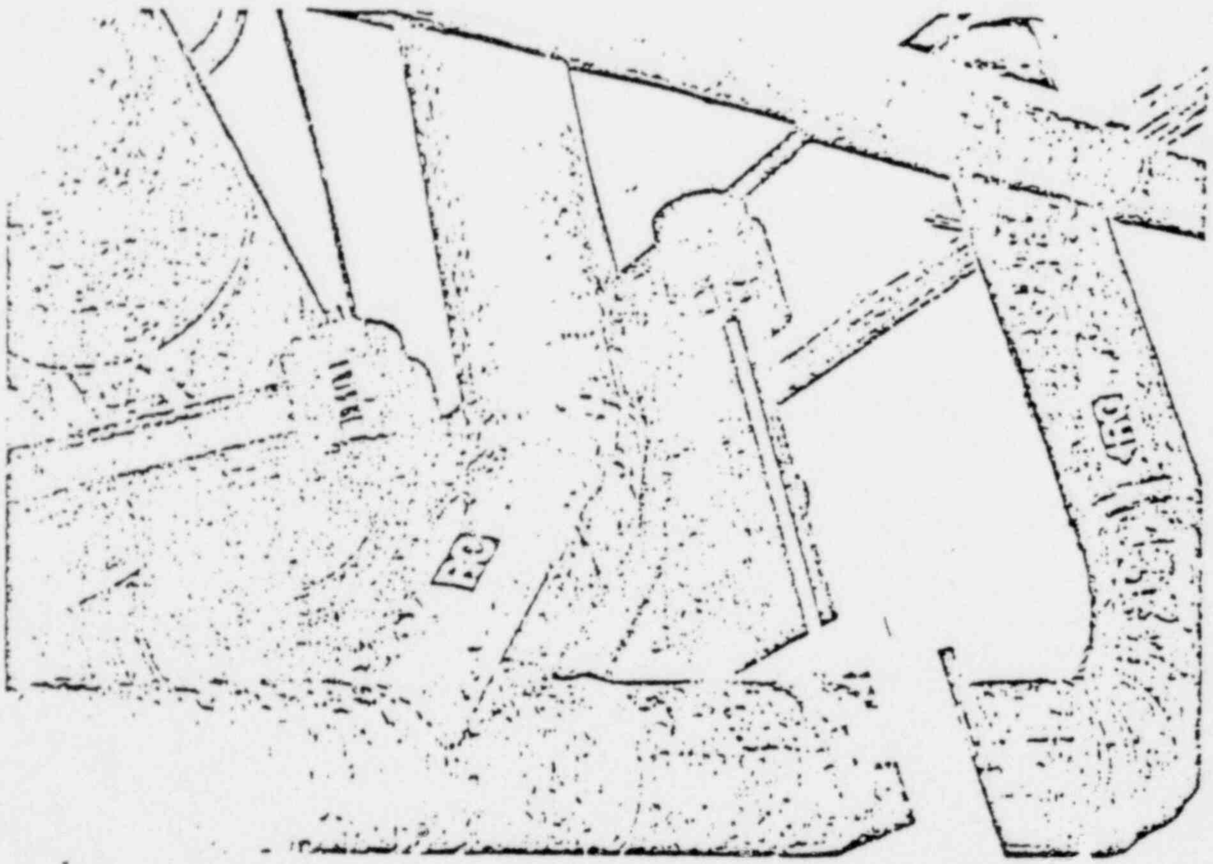


Figure 12 - Support R4 (2)

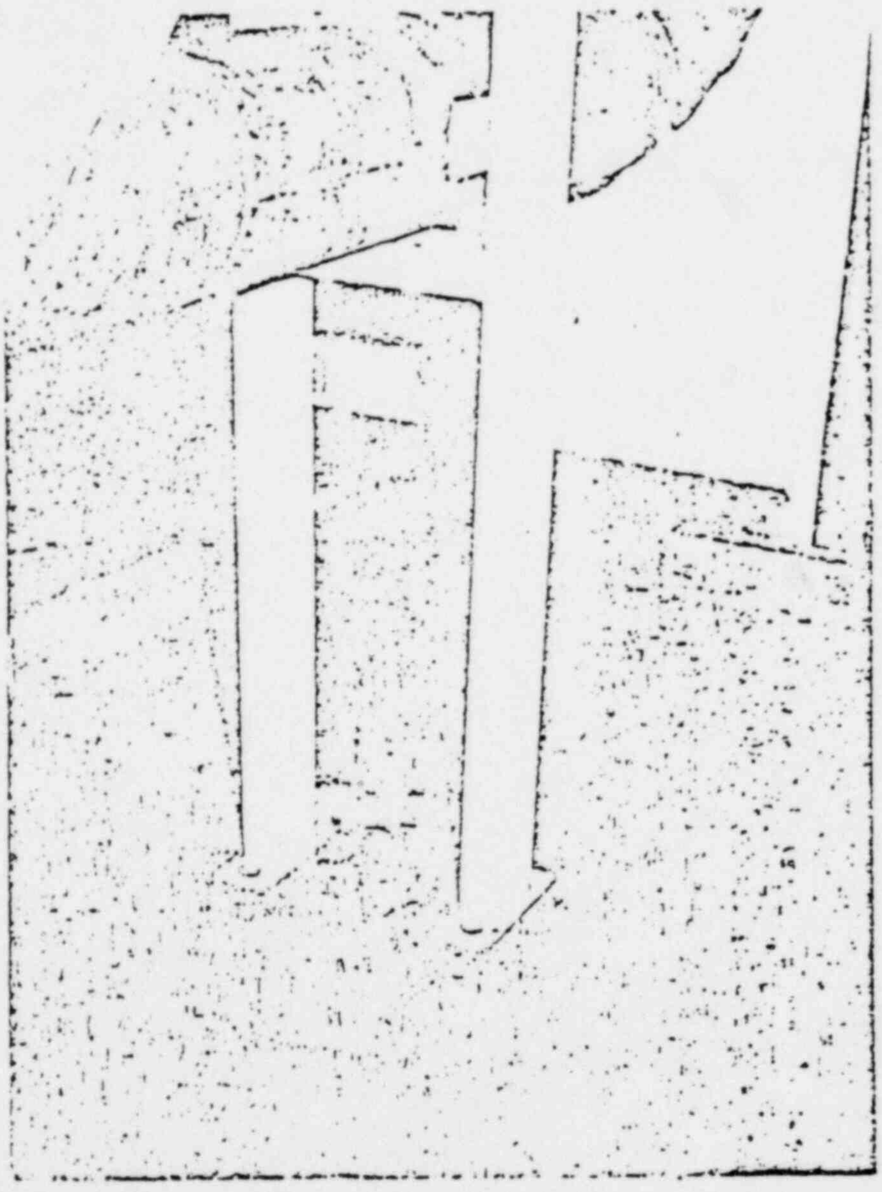
POOR ORIGINAL

BEZNAU - UNIT N° 1 (NC

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE



POOR ORIGINAL

Figure 13 - Support R4 (3)
Attachment to floor
Concrete damage (back of the restraint)

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE.

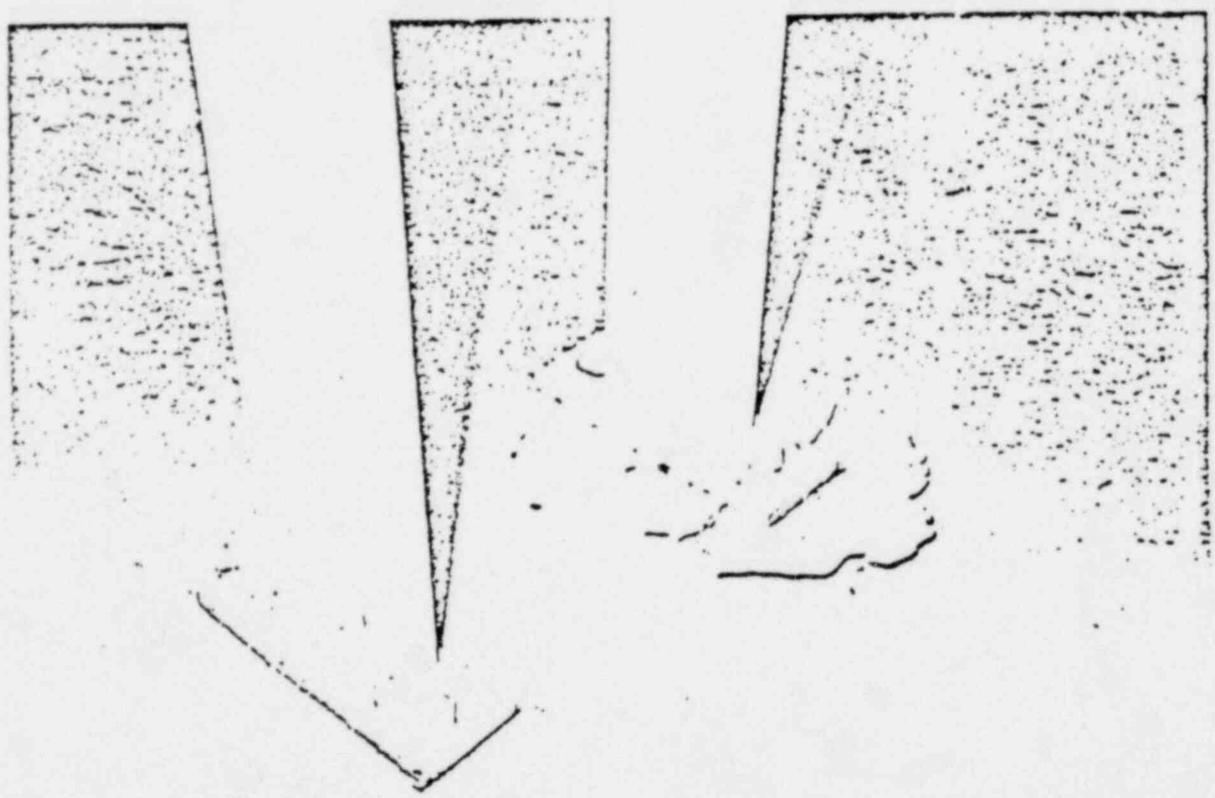


Figure 14 - Support R4 (4)

Detail of concrete damage.

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE-INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE.

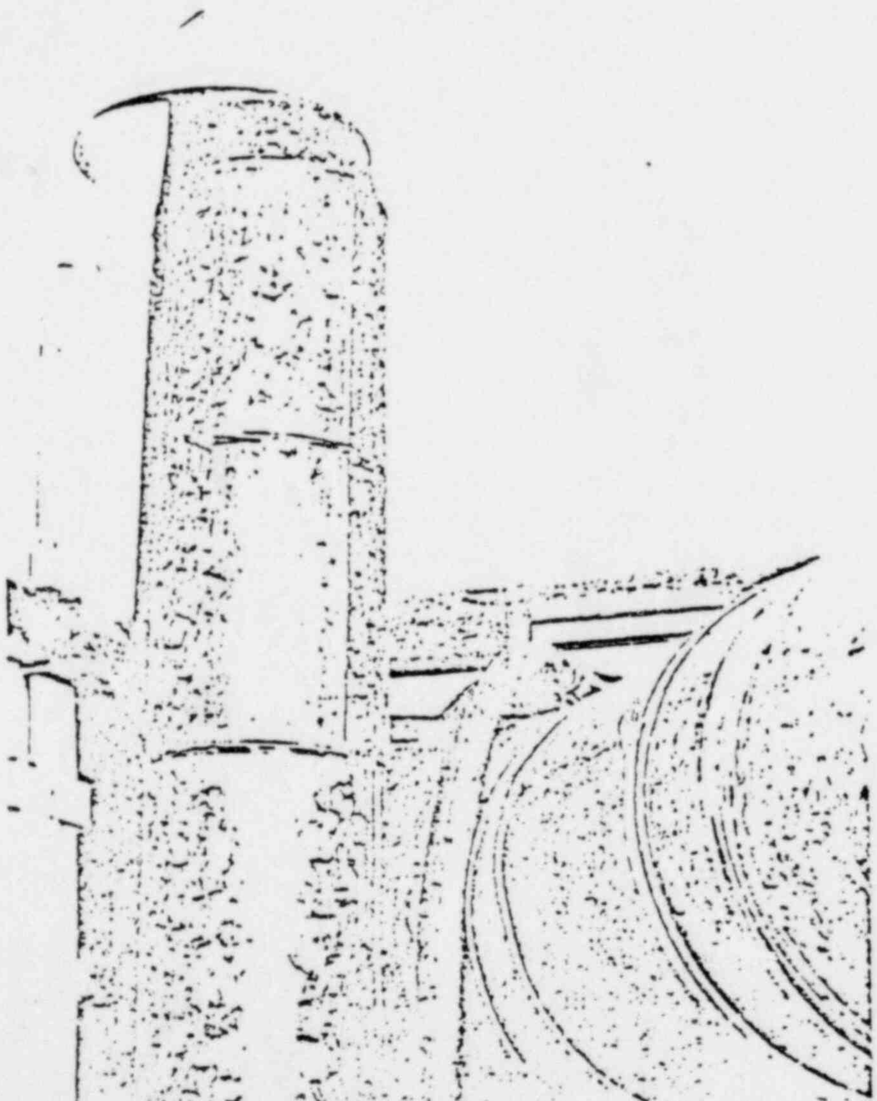


Figure 15 - Ceiling Penetration (1)

POOR ORIGINAL

BEZNAU - UNIT N° 1 (NOK)

STEAM DUMP FAILURE INCIDENT

Aug. 21, 74

PRESSURIZER RELIEF LINE.

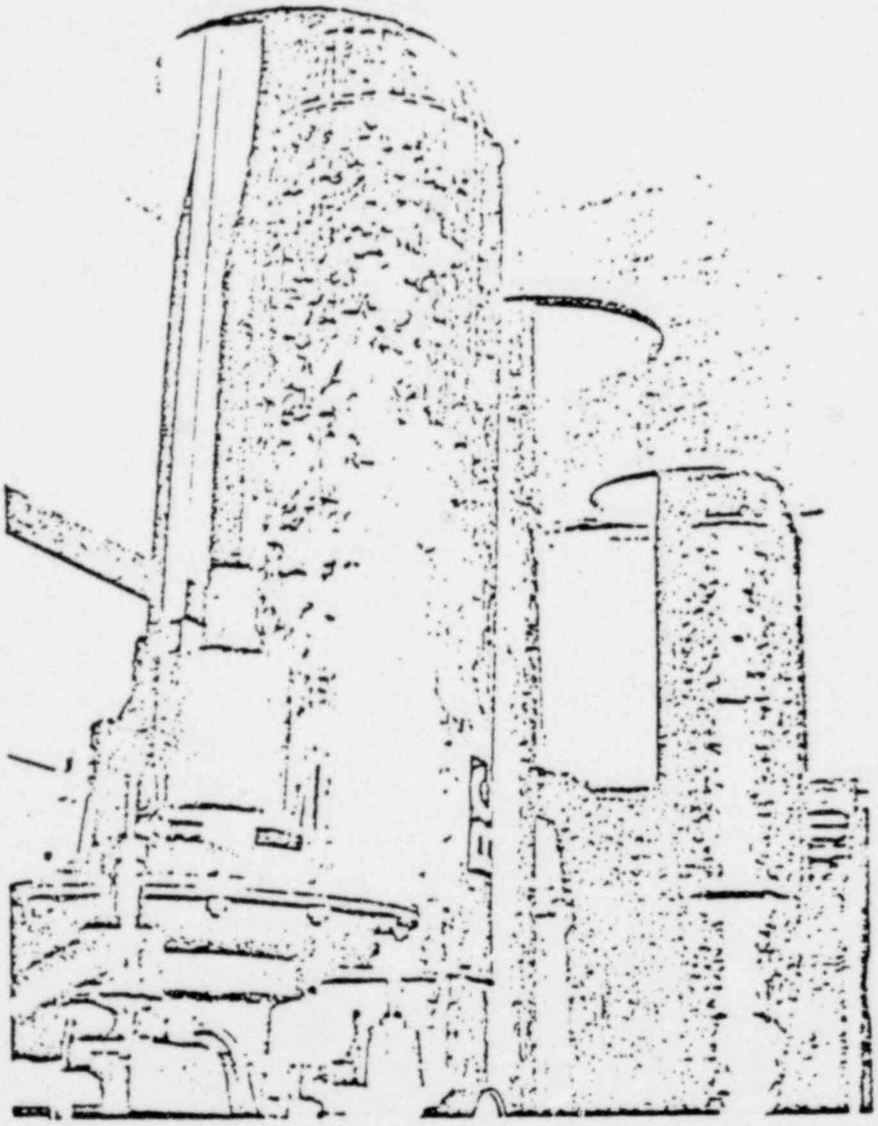


Figure 17- Ceiling Penetration (3)

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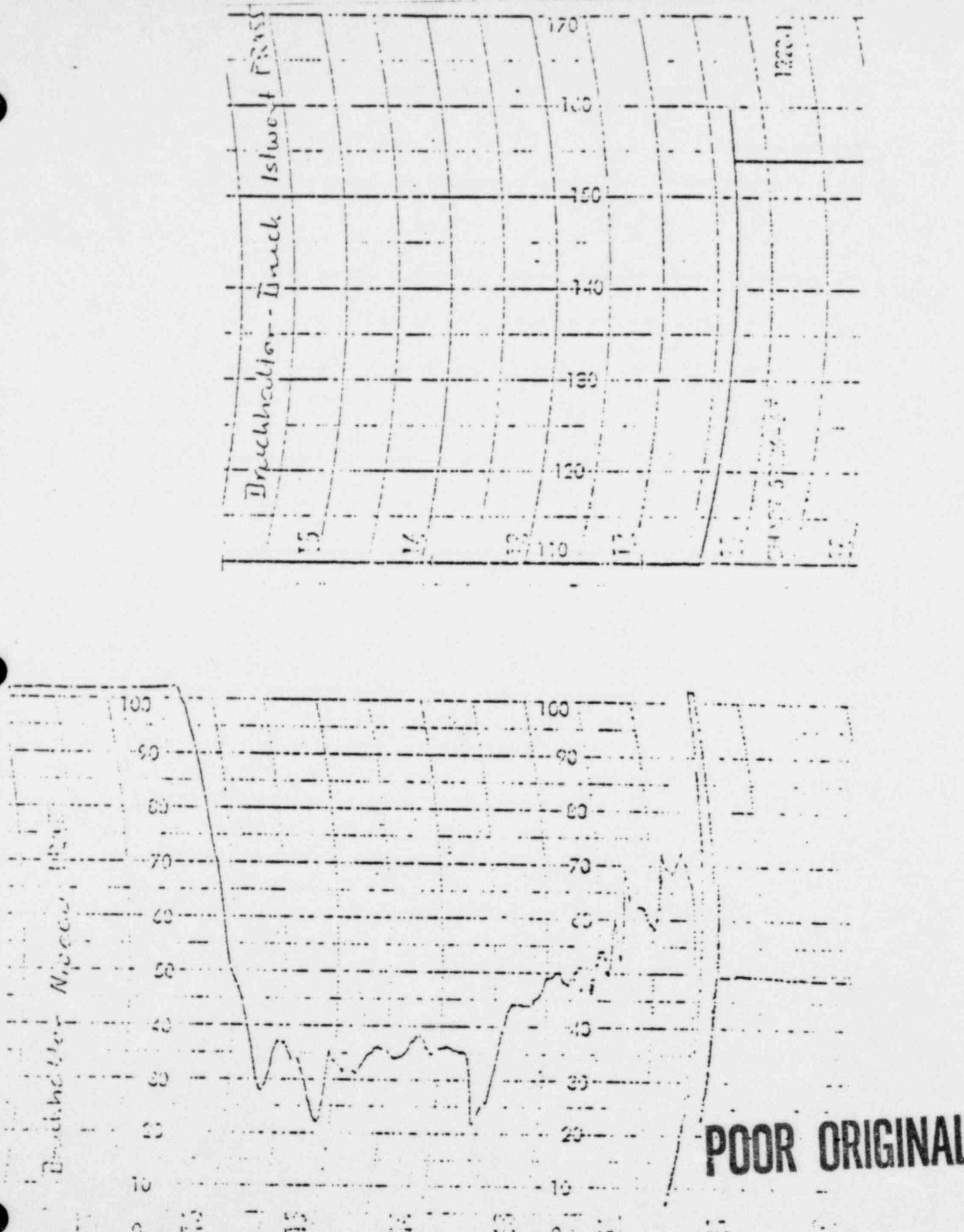

Figure 18 : Record of Pressurizer Pressure and Level Transients following incident initiation.

NOK REPORT ON BEZNAU ACCIDENT OF AUGUST 20, 1974

1. TRIP TG-1/REACTOR TRIP/SI/

On August 20, 1974 at 11:20 a.m. a trip on turbine TG-1 occurred resulting to high bearing and casing vibrations (Bearing 6:60)

At trip time, generator 2 was delivering about 140 MVar. Resulting from a failure of the steam dump system to operate, with the consequence that the relief valve did not open. That resulted in a rapid rise of coolant temperature, steam pressure and pressurizer level and pressure.

At 160 bar of pressure in the primary, the pressurizer pressure relief valves opened, lowering rapidly the pressure in the primary. About 10 seconds after valve opening, the pressure had reached such a low level that the pressurizer pressure relief valves were reactuated to close. Due to a disturbance, valve PCV-456, failed to close, resulting in a lowering of RCS pressure up to 100 bar after about 1 minute. Reactor tripped resulting from a low pressure signal (126.5 bar).

Due to the opening of the pressurizer relief valve, the pressure in RCS dropped to about 70 bar, corresponding to a saturation temperature of 284°C. Consequently, steam appeared in the primary hot leg, filling the pressurizer.

Two or 3 minutes after trip, the operator recognised the failure of the relief valve and isolated it with the power operated valve 531. The water level began to drop, and 11 minutes after trip, automatic SI was initiated by low pressure and level in the pressurizer.

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SI systems worked normally and about 40 litres per second of water was spilled through the four SI pump nozzles into the primary, causing a rise of pressure to 110 bars and a further rise of level to 70 %. The SI pumps were then turned off and the power operated valves of the spray pipings were closed.

From that moment on, the pressurizer level could be controlle through charging pumps and release of steam, assuming the primary to cool down.

About 3 minutes after trip, the containment pressure alarm signal was actuated because of too high pressure, and 1 minute later the high activity alarm. Maximum pressure in containment reached 100 mbar over normal. The operators activated the containment fan coolers. Since several safety alarms of the pressurizer relief tank were on, it was quickly assumed that the rupture disc was broken and that the discharge channel was defectuous. After TG-1 trip, due to steam dump failure, steam pressure rose to 66 bar.

The turbatrol of TG-2 was actuated as an emergency after TG-1 trip. TG-2 was unregular in behaviour, and the position of the control valve remained constant during the pressure transient. The performances of TG-2 rose to about 214 MWe due to higher steam pressure (rise from 52 bar to 66 bar).

After TG-2 trip, following reactor trip, steam pressure rose to over 70 bar, actuating the safety valves and thus lowering pressure to about 65 bar.

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2. CHRONOLOGICAL SEQUENCE OF EVENTS

August 20, 1974

2.1. Reactor Trip

Beginning of incident	11 h 20' 12"
TG-1 main breaker off	
Pressurizer pressure low-trip	39,7" later
Reactor trip breaker open	39,8" later
TG-2 main breaker off	40,3" later
SI actuation (pressurizer pressure and level low)	11'55,9" later

2.2. Events as Registered on Alarm Typewriter

TIME		
11:15	TG-1 power high	135,5 MVar
11:20	Allowable oil pressure of TG-1 too low	
11:20	Pressurizer pressure high.	158.2 bar
11:20	Pressurizer pressure high.	159.9 bar
	Reactor Trip.	
11:21	Tavg RCS-A high	302.2°C
11:21	Steam pr. upstream of TG-1 stop valve high.	66.3 bar
11:21	Tavg RCS-A high	305.2°C
11:21	SG-A steam pressure high.	67.3 bar
11:21	SG-B steam pressure high.	67.2 bar
11:21	Steam pr. upstream of TG-1 stop valve.	77.6 bar
11:21	SG-A steam pressure high.	73.3 bar
11:21	SG-A steam pressure high.	65.4 bar
11:22	Safety oil pressure of TG-2 too low.	
11:22	Tavg RCS-A	285.2°C

TIME		
11:23	Steam pressure upstream of TG-2 stop valve.	68.1 bar
11:23	Pressurizer relief tank temperature high.	62.8°C
11:24	Pressurizer level	79 %
11:24	Pressurizer level	88 %
11:24	Containment pressure high	1.1 bar abs
11:24	Pressurizer relief tank level low.	20.2 %
11:24	Pressurizer relief tank pressure high.	0.59 bar
11:25	Pressurizer relief tank pressure	0.15 bar
11:25	SG-A+B steam pressures normal.	63.7 bar
11:25	Containment activity high	17.3 mr/h
11:26	Loop B RCS flow low.	88 %
11:27	Containment air temperature high	53.4 °C
11:32	Pressurizer level low.	6.8 %
11:32	Pressurizer level normal.	18 %
11:33	Surge line temperature too low.	271.1°C
11:34	Pressurizer level high.	58 %

2.3. Sequence of Events for Pressurizer and Pressurizer Relief Tank

TIME		
11 h 20'	11.1"	Pressurizer pressure above control range.
	11.9"	Pressurizer relief valve.
	22.8"	Pressurizer relief tank pressure high
	23.0"	Pressurizer relief valve locked
	23.0"	Pressurizer pressure normal
	23.1"	Pressurizer relief tank level high
	24.2"	Pressurizer level high.
	33.0"	Pressurizer relief tank pressure too high.
	35.0"	Pressurizer pressure under normal.

TIME

11 h 21' 00.4"	Pressurizer pressure low - Trip.
01.2"	Pressurizer pressure low - SIS unlocked.
05.1"	Pressurizer relief tank level high.
13.5"	Pressurizer pressure low - SIS unlocked.
11 h 23' 27.6"	Pressurizer level high - 1 channel tr:
43.3"	Pressurizer relief tank level too hig:
43.5"	Containment pressure too high.
47.1"	Pressurizer relief tank level low.
11 h 24' 29.4"	Pressurizer relief tank pressure norma
51.2	Containment temperature high.
11 h 25' 17.8"	Containment activity high.

3. ANALYSIS OF THE CAUSES OF THE INCIDENT

TG-1 tripped due to high casing vibrations, especially in casing 6. It had already been noticed that TG-1 was sensitive to shocks. At the moment of incident, TG-1 was set to function under maximum effort, so that it could support a maximum of vibrations.

The trip is not unfamiliar and would not have affected the primary if steam dump had normally been actuated.

An inspection of containment after primary had cooled down, showed that the yoke between the PCV-456 valve housing and air engine was broken, and probably due to a dynamic effort on the piping at opening of the valve.

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Consequently, the valve failed to close and initiated a rapid fall of pressure in primary. The pressurizer relief tank rupture disc broke, due to a prolonged surge of primary coolant in the tank. Items 2 and 3 show the disc broke when the relief valve had already closed.

WATER COLLECTED IN CONTAINMENT SUMP

Regen. hold up water Tank A 38 % - 100 %	= 9.8 m^3
Regen. hold up water Tank B 16 % - 36 %	= 3.2 m^3
Total quanlity of water collected	=13.0 m^3
Pressurizer relief tank 80 % - 19 %	=11.2 m^3
Water out of system.	= 1.8 m^3

Since no further damage was noticed in containment, it could be assumed these 1.8 m^3 of water were blown out.

4 1. Thermal Stresses in RCS

Beside a rapid water temperature rise of about 6°C after TG-1 tripped, a rapid primary pressure rise from 154 bar to 160 bar, there was also an important temperature transient in area of SI nozzles. However, since the reactor's main pumps operated all the time, thus mixing cold spray water with hot coolant, it can be assumed that other components didn't undergo high temperature gradients. Furthermore, nozzle temperature and stress remained within design limits.

4.2. Damages to Relief Systems

During inspection in containment after cooling of primary, the following damages in the pressurizer relief systems were observed :

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- relief valve PLV 456 : Mechanism broken on both sides and bent spindle.
- One anchor point of the relief system piping after valve was loose.
- Relief tank pressure disc broken.

Further damages in containment were not noticed.

It must be said that the relief tank is not designed to accept steam from the pressurizer for a prolonged time. The damages to the relief valve is therefore a direct cause to the breaking of the rupture disc.

4.3. Turbines

TG-1

The cause of vibrations to the casing are most probably the stresses and shocks. The P signal from hydrogen seal oil system is due to casing vibrations.
Damages to the seal or casing are most improbable.

TG-2

The oscillation from 172 MWe to 110 MWe, and then to 215 MWe suggested that the bolts of the high pressure cylinder were loosened and had lost some of their tension.
A too small stress was noticed, due to leakage of the seals of the high pressure cylinder. Due to too high rotational momentum at 215 MWe, the coupling between turbine and generator was closely controlled.

5. When reviewing the sequence of events, the failure of two systems, namely the steam dump and the pressurizer relief system, we came to the conclusion that it did not bring to an uncontrolable nor a damaging situation. During the incident, no activity (in gas or liquid form) in the surrounding area reached an uncontrollable level.

The generator safety valves maintained the steam pressure within allowable limits. The SIS brought back the primary to a safer pressure, allowing normal cooldown conditions.

6. PROPOSAL FOR MODIFICATIONS

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6.1 Control of generator 1

Generator 1 reaching rapidly to casing vibrations, it will be tried to see if the regulator can be modified in order to have a quick action.

6.2. Pressure Regulator

Tests will be made to see if the first row of impellers in the pressure regulator of the turbine must not be reviewed in order to limit power to 190 MWe.

6.3. Steam Dump System

a) Revisions and calibrations should be made in steam dump system (before opening of steam dump valve.)

b) Studies will be made, to make periodic controls of steam dump while in operation. It should help to insur better safety limits (for example : unwanted opening of steam dump valve).

c) A control type writer linked to the steam dump will be installed in order to control the opening of steam dump valves and to check the good working of oil pumps.

6.4. Pressurizer Relief System

The first measure to be taken, is to repair the damaged valve, the piping supports and review boltings.
The pressurizer relief tank rupture disc must be replaced.
With these repairs start-up should be possible.

To see how the relief system piping can be better secured and how shock at opening of relief valve can be avoided are further measures to be taken.

D. Ross

ERhit 1154

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

September 27, 1979

MEMORANDUM FOR: ALL B&O PERSONNEL

FROM: C. J. Heltemes, Group Leader, B&O Task Force

SUBJECT: INCIDENT AT BELGIUM DOEL 2 REACTOR

The attached memorandum and report describe a rather large, sudden steam generator tube rupture at the Doel 2 nuclear power plant in Belgium. This is provided for your information and use in the evaluation of potential generic and plant specific failure modes.

CJ Heltemes Jr
C. J. Heltemes, Jr., Group Leader
Bulletins and Orders Task Force

Attachment:
As stated

POOR ORIGINAL

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

SEP 13 1979

MEMORANDUM FOR: H. R. Denton, Director, NRR
E. G. Case, Deputy Director, NRR
D. Ross, Deputy Director, DPM
R. Mattson, Director, DSS

FROM: Darrell G. Eisenhut, Acting Director
Division of Operating Reactors

SUBJECT: INCIDENT AT BELGIUM DOEL 2 REACTOR

I response to our following up on a rather large, sudden steam generator tube rupture at the Doel 2 nuclear power plant in Belgium, we have received the attached report. You may find this incident particularly interesting since the unit underwent a transient where pressurizer level apparently went offscale high. Strip chart recordings of the event are enclosed.

We hope to be obtaining more information on this event in the near future.

Darrell Eisenhut
Darrell G. Eisenhut, Acting Director
Division of Operating Reactors

Enclosures:
As Stated

cc: S. Hanauer
F. Schroeder
B. Grimes
P. Check
G. Lainas
S. Levine
V. Stello
W. Russell

Rick 7/27

CENTRE D'ÉTUDE DE L'ÉNERGIE NUCLÉAIRE

C.E.N. / S.C.K.

ÉTABLISSEMENT D'UTILITÉ PUBLIQUE

Veuillez adresser votre réponse
en deux exemplaires aux
LABORATOIRES DU C.E.N./S.C.K.
Boeretang 200 B-2400 MOL

Tel. (014) 31 18 01
Telex SCKCEN-Mol 31922
Adr. télégr. Centratom Mol

SEP 3 1979

Mr. Joseph D. LAFLEUR, Jr.
Deputy Director
Office of International Programs
UNITED STATES NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555
U.S.A.

MOL, le 21.08.79.

V/lettre V/ref. N/ref.
Centrale BR3
FM./mb
5.5126/71

Dear Dr. LAFLEUR,

As a first answer to the telex of Mr. H.J. FAULKNER NRC-BHDA, dated 8.8.79, I send you here enclosed a report describing the steam generator leak incident at the Unit 2 of the Doel nuclear power plant.

This report has been transmitted to me by "Tractionel Engineering", a division of the compagny "Société de Traction et d'Electricité" in Brussels ; as you most probably know, this division is playing the role of engineering office for the benefit of the Doel plant operator compagny (EBES).

I hope you will find in this report satisfactory answers to all your questions ; do not hesistate to ask for eventual additional informations.

Yours sincerely,

F. Motte

F. MOTTE
BR3 Plant Superintendent.

Enclosure : "Report on the incident at Doel 2 nuclear power plant
Severe leakage in steam generator B on June 25, 1979".

20.07.79

SEP 3 1979

REPORT ON THE INCIDENT AT DOEL 2 NUCLEAR POWER PLANT

SEVERE LEAKAGE IN STEAM GENERATOR B ON JUNE 25, 1979.

1. STATUS OF THE POWER PLANT AT THE MOMENT OF THE INCIDENT

The primary system was being heated up after repair works at the actuation system of the main steam valve.
At the moment of the incident, temperature in the primary system was ± 255°C (refer to point A on Fig. 1 & 2) and pressure had reached its rated value of 157 kg/cm^2 (refer to point A on Fig. 3 & 4).
The reactor was subcritical with all rods in.
Secondary pressure in the steam generators was ± 45 kg/cm^2, the saturation pressure corresponding to 255°C (refer to point A on Fig. 6 & 7).
For some time, A-loop steam generator had shown a low activity value along the secondary side (below admissible limits) that indicated a small leakage.

2. SEQUENCE OF THE EVENTS (refer also to various computer data given in attachment)

2.1. Initiating phase

About 7:20 PM, a quick pressure decrease is recorded in the primary system (about 2 kg/cm^2 per minute : see Fig. 4), which results in accelerating the operating charging pump. A second charging pump is started manually. The letdown

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Annexe
425

station of the CV system closes automatically. It is confirmed that the relief valves are closed and their isolation valves are preventively closed. The level in the pressurizer quickly decreases (see Fig. 5) and the electrical heaters are automatically disconnected.
At the same time, a quick level increase is recorded in B-loop steam generator (see Fig. 7 point B). The activity measurement channels of the blowdown system record a maximum value.
The combination of all those signals indicates a severe leakage in B-loop steam generator. The faulted steam generator is then immediately completely isolated along the steam side and the discharge valve to the atmosphere is set at maximum pressure.

Meanwhile the third charging pump is started (was set apart to be maintained), but the three charging pumps are not sufficient to compensate the loss of fluid in the steam generator. Indeed, the CV tank is readily empty and the charging pumps are automatically supplied from the 2R11 refuelling water storage tank. To increase the subcooling primary pump B is stopped and letdown starts through A-loop steam generator (see Fig. 3, point B).

2.2. Actuation of safety injection

About 20' after the incident started, the threshold pressure (118.5 kg/cm^2) to actuate the safety injection is reached. The emergency diesels start within the required time lapse but are not necessary. Phase A isolation and ventilation isolation of the reactor building are achieved. The vital components not yet in operation are started.

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When reaching the 108 kg/cm^2 value, all HP SI-pumps discharge into the primary system, and the pressure decrease is stopped (see Fig. 3, point C).
To prevent the secondary pressure in the faulted steam generator from reaching the opening pressure of the safety valves, the primary pressure is successfully decreased (see Fig. 3, point D) through maximum spray in the pressurizer (re-start of primary pump B and use of both spray lines). During this phase, the level in the pressurizer quickly increases and it fills up completely (see Fig. 5). Spray is temporary stopped and pressure stabilizes at zero flow pressure of HP SI-pumps.
The automatically started auxiliary feedwater supply results in a pressure decrease in B-loop steam generator (see Fig. 7, point C). The auxiliary feedwater supply pump of the faulted disconnected steam generator is locally stopped and isolated (Fig. 7, point D). This cannot be performed from the control room since the SI signal still prevails. The auxiliary feedwater supply tank is filled up from Doel 1.

2.3. Cancelling of SI-signal

Pressure decrease was now mandatory :

a) to avoid the opening of safety valves of the faulted steam generator.
b) to start, as soon as possible, the shutdown cooling system (low pressure circuit 1) to stop the letdown of slightly contaminated steam through the A-loop steam generator.

First,the safety injection signal had to be cancelled. This had to be performed more than once (each time requiring 5 minutes interval) because of a relay fault.

After definitively cancelling the SI-signal, two HP SI-pumps are stopped and soon thereafter a third one (Fig. 3, point F). While considering the subcooling margin, the last HP SI-pump is stopped. Pressure successively decreases to reach $\pm$ 65 kg/cm^2 (Fig. 3, point H) (saturation pressure is $\angle$ 15 kg/cm^2 at that moment).

It is then tried to initiate the CV-discharge line, but valves do not open. Some time goes by before the reason therefore is determined. Due to phase A isolation there is no longer a compressed-air supply in the reactor building. After re-opening the compressed-air supply line the discharge line is opened (Fig. 3, point I). Pressure decreases, first quickly, then slower.

The loss of compressed-air supply has also resulted in the closure of CC-valves to the primary pumps. The pumps have run for a long time without cooling of the thermal shield, however without alarm temperatures were reached.

2.4. Initiation of the residual heat removal system

As the CV-system permittted only a slow pressure decrease, the interlock, which maintains the isolation of the RHRS up to a pressure of 28 kg/cm^2, has been bypassed at 31 kg/cm^2 There was indeed a sufficient margin compared to the design pressure of the system (42 kg/cm^2). Thanks to this operation the letdown through A-loop steam generator could be stopped earlier and the discharge of slightly contaminated steam could be reduced (Fig. 3, Point J).

2.5. Further sequences

The abovementioned operation allowed a primary pressure decrease below the value of secondary pressure in the faulted B-loop steam generator. The secondary level decreases, which creates a dilution risk. The boric acid concentration is controlled every half hour (stabilized howerver at ± 1500 ppm).

Thanks to the cooling down, pressure decreases slowly in B-loop steam generator and reaches a value lower than the primary pressure. From this moment on, attention is paid to always maintain the primary pressure higher than that in the steam generator.
Despite the cold water so discharged in the steam generator, pressure goes on decreasing slowly (due to the presence of a warm water film at the water surface).
As the level of water in the steam generator approaches the upper limit of the broad level measurement pressure is sufficiently low (± 12 kg/cm^2) to inject nitrogen.
The secondary drain line is coupled with system B for liquid waste, and the steam generator discharges into it through the nitrogen pressure.
The nitrogen is only slightly contaminated after this and can be discharged via the annulus between primary and secondary containments.

2.6. Comments and conclusion

The incident has been handled as prescribed and no damages have occured to the envirorment or the installation.
The procedures have to be reviewed considering the following :

- cancelling of phase A isolation to restore compressed air supply in the reactor building.

Attachment 1 - Computer data

1. Initiating phase

19 21'06" pressurizer pressure below reference pressure

19 22'51" demand for charging pump higher speed

19 23'31" disconnecting pressurizer heaters by low level

19 23'32" CV letdown station valves closed

19 25'42" closing of isolation valves of relief valves and spray valves

19 26'14" low pressure in primary system

19 30'30" very low pressure in pressurizer

10 30'30" high level in B steam generator

19 38'32" B primary pump disconnected

2. Safety injection phase

19 40'18" low pressure in pressurizer

19 40'19" safety injection through low pressure in pressurizer

19 40'19" diesels started

19 40'19" reactor building ventilation isolation

19 40'20" phase A reactor building isolation

19 40'24" actuation signal HP SI-pumps

19 40'33" HP SI-valves opened

19 43'28" very large auxiliary feedwater flow to A SG

19 44'39" very large auxiliary feedwater flow to B SG

19 53'12 auxiliary feedwater supply pump B disconnected

19 56'37" very low level in auxiliary feedwater supply tank

19 57'11" pressurizer level normal

19 57'29" pressurizer heaters re-started

19 58'48" high level in pressurizer

3. SI-signal cancelling phase

20 00'15" automatic starting signal of diesels cancelled and SI-pumps starting signal cancelled

20 00'21" back to SI

20 03'24" LP compressed air in reactor building

20 05'59" safety injection ordered

20 06'05" safety injection

20 10'59" reactor building ventilation isolation ordered

20 21'15" HP SI-pump B disconnected

20 25'22" HP SI-pump A disconnected

20 38'33" valve CC 096 closed

20 40'25" valve CC 099 closed

20 48'54" compressed air supply to reactor building restored

20 49'00" primary pumps CC-valves re-opened

4. Actuation of RHRS

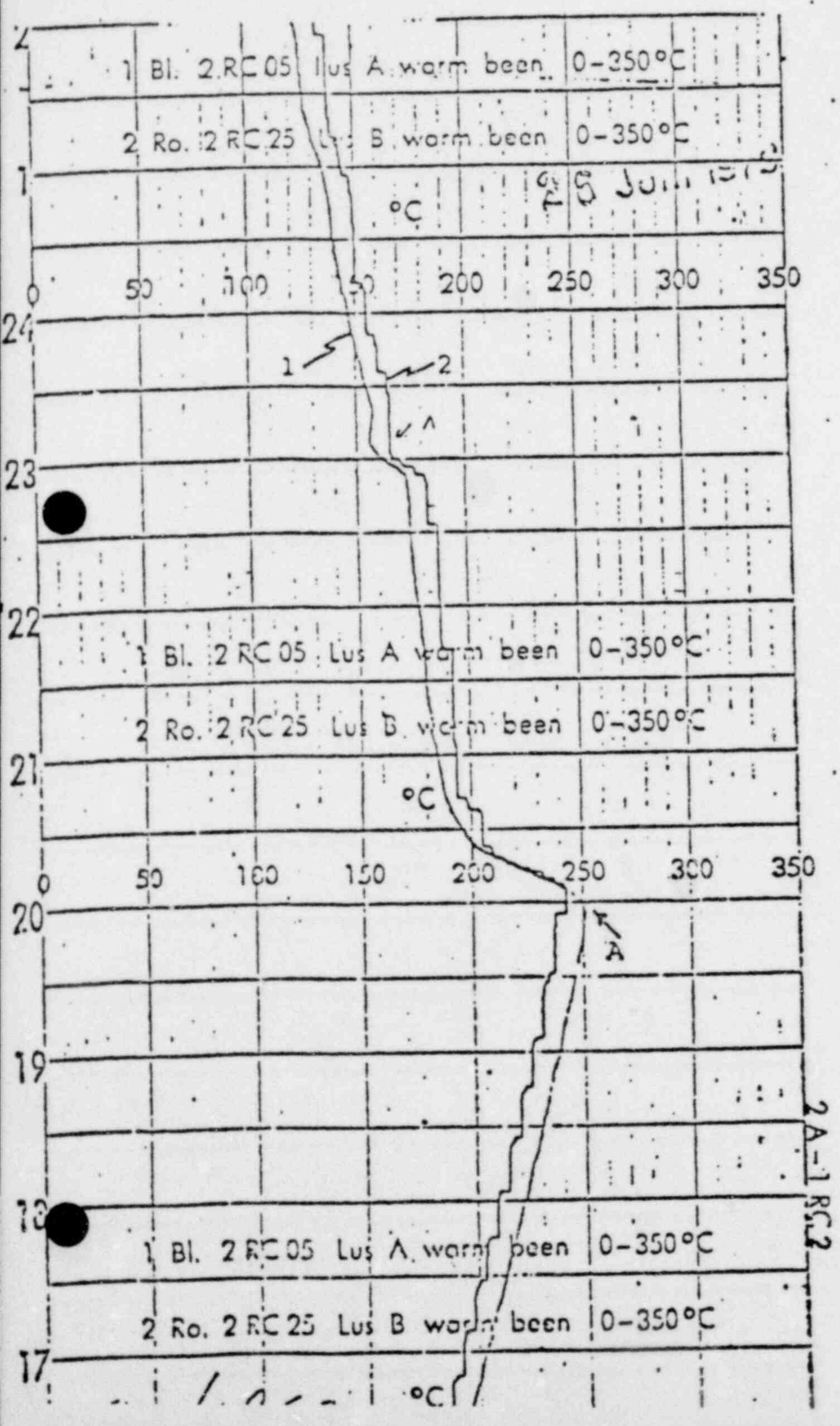
22 35'54" valve RC 003 opened

...ture no. 6

...R WARM BEEN

FIGUUR 1

: ...A - 1 RC 2
Loop A - hot leg
05 lus A warm been — 0 - 350°C
loop B hot leg
25 lus B warm been — 0 - 350°C



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Pressurizer level

PEIL R 2 FIGUUR 5

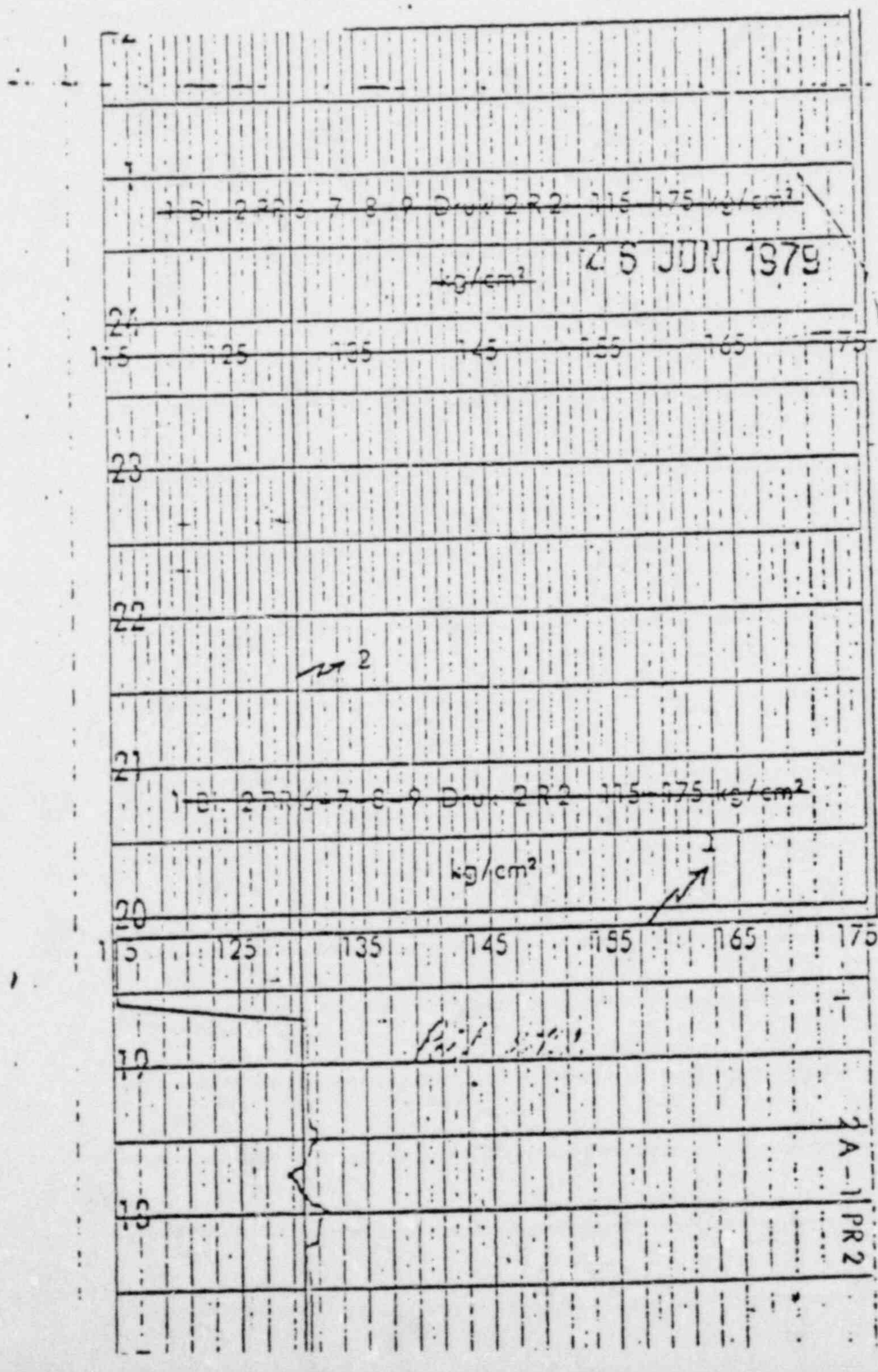
Schrijver : 2 A - 1 PR 1 Pressurizer level

1. Bl. L. PR 11 - 12 - 13 Peil R 2 0 - 100 %

Reference level

2. Ro. L. Ref. Ref. Peil R 2 0 - 100 %

Meting in dienst spoor 1 :



verkeerde schaal vervangen door 0 → 100 %

wrong scale Replaced by 0 - 100 %.

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SG B - level and pressure

SG. B PEIL - DRUK FIGUUR 7

Schrijver : 2. A - 4 FW MS 5

SG B level

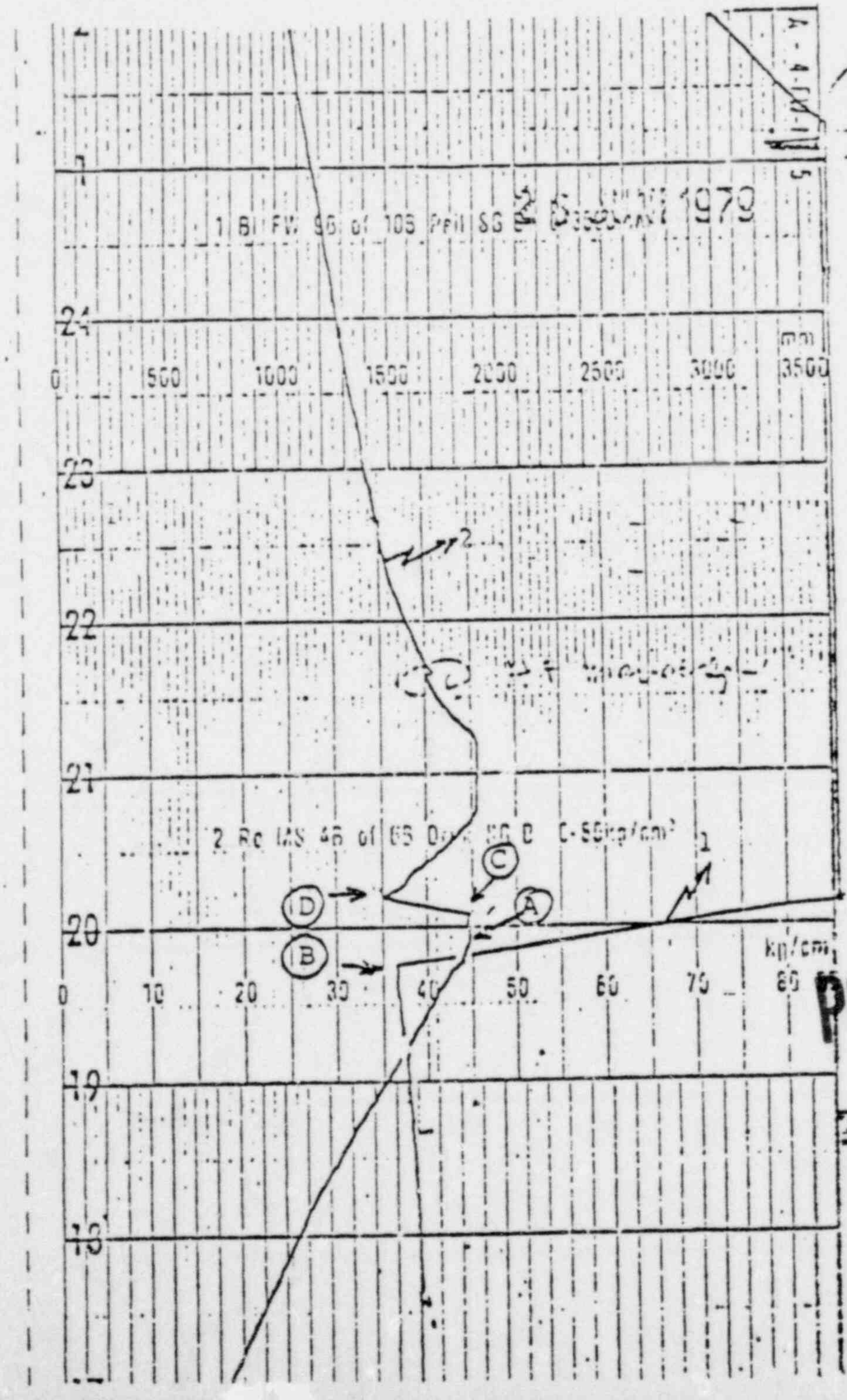
1. Bl. L. FW 9 B - 10 B Peil SG B 0 - 3500 mm

SG B pressure

2. Ro. P. MS 4 B - 6 B Druk SG B 0 - 85 kg/cm2

Meting in dienst spoor 1 :

. spoor 2 :



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NRC BHDA
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ICS IPMIIHA IISS
IISS FM WUI 30 0640
PMS ORY COMMISSION WASHINGTON DC
UWE9303 3E32221TG3/229
UWWA CO BEAN 132
ANTWERPEN TELEX 132/125 30 1007 P1/50
DR JOSEPH D LAFLEUR JR
DEPUTY DIRECTOR
OFFICE OF INTERNATIONAL PROGRAMS
U STATES NUCLEAR REGULATORY COMMISSION
WASHINGTONDC(20555)
COMPLEMENTARY TO MY LETTER REF 5.5126/71 OF AUGUST 21 PLEASE
FIND HEREAFTER SPECIFIC ANSWERS TO THE FOUR QUESTIONS RAISE BY
YOUR DR FAILKNER ON THE DOEL 2 STEAM GENERATOR INCIDENT
1. THE MAGNITUDE

COL (20555) 5.5126/71 21 2 1.
3/229 DR JOSEPH D LAFLEUR JR P2/50
OF THE LEAK WAS ESTIMATED AS ABOUT 30 TONS/HOUR AND
DEVELOPPED RADPIDLY
2. THE LEAK IS LOCATED ON THE TOP OF PIPE NR 1/24 OF STEAM GENERATOR
B DOEL 2 IN THE EXTRA-DOS OF THE U-BEND
3. SUSPECTED CAUSE STRESS-CORROSION DUE TO OVALIZATION
4. DENTING MAXIMUM 450 MICROMETER
NO FLOW
COL 30 2. 1/24 2 3. 4. 450

3/229 DR JOSEPH D LAFLEUR JR P3/25
SLOT DEFORMATION AT ALL NO TUBE WALL THINNING FOUND
THESE ANSWERS WERE FORMULATED BY THE DOEL 1 AND 2

POOR ORIGINAL

PLANT SUPERINTENDENT
YOURS SINCERELY
F MOTTE
COL 1 2
RETR MSG
NNNN
NN
WU TWX WSH

NRC BHDA

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Exhibit 1155

OCT 2 0 1977

Note to: Karl V. Seyfrit, Assistant Director, Division of Reactor Operations Inspection, IE

Subject: DAVIS-BESSE 1 ABNORMAL OCCURRENCE (9/24/77)

Some areas of interest to us that are appropriate for the IE formal report are:

1. Potential for, and core cooling consequences of, insulation debris inside containment after a LOCA. If large pieces could break off, could they get to, and block the sump?

2. The operator's role in participating in the event should be related. For example, the manual actions associated with the control of level in SG #2 should be described. The operator's decision to secure HPI flow based on pressurizer level indication should be explained.

3. The dynamic effects of vapor formation in the reactor coolant system during the transient (where and when it occurred, RC pump cavitation effects, RC pump seal effects, etc.) should be described.

4. Adequacy of AFW capacity with regard to this transient are of interest. For example, evaluate the observed primary side heatup against the design capability of one AFW train. Also, the adequacy of the AFW actuation setpoint (SG level) should be examined against the number of cyclic stresses allowed over the life of the plant.

D. F. Ross, Jr., Assistant Director
for Reactor Safety
Division of Systems Safety

cc: G. Mazetis
T. Novak

Contact
G. Mazetis, NRR
Ext. 27341

POOR ORIGINAL

Exhibit 1156

FEB 0 6 1978

MEMORANDUM FOR: D. B. Vassallo, Assistant Director for LWR's, DPM

FROM: D. F. Ross, Jr., Assistant Director for Reactor Safety, DSS

SUBJECT: TMI-2, INPUT TO SER SUPPLEMENT NO. 2

Plant Name:	TMI-2
Docket No.:	50-320
Milestone No.:	27-21
Licensing Stage:	OL
Responsible Branch and Project Manager:	LWR-4 H. Silver
Systems Safety Branch Involved:	Reactor Systems
Description of Review:	Input to SER Supplement No. 2
Review Status:	Complete

The Reactor Systems Branch has prepared the attached input for SER supplement number 2. This completes the RSB review of Three Mile Island Unit 2. The following topics are addressed.

5.2.2 Overpressure Protection during Startup and Shutdown*

6.2 Net Positive Suction Head Assessment

6.3.3 Makeup Tank Isolation*

15.2.2 Steamline Break Analysis*

15.2.2.1 Secondary System Modification*

15.2.2.2 Long Term Cooling following a Steamline Break*

15.2.8 Feedwater Line Break

Note: Items marked with * involve limits on plant operation

D. F. Ross, Jr., Assistant Director
for Reactor Safety
Division of Systems Safety

Enclosure:
SER Input

773430394

POOR ORIGINAL

cc: S. Hanauer
R. Mattson
D. Ross
S. Varga
H. Silver
T. Novak
S. Israel
J. Watt

Distribution
Docket File
NRR REading
RSB Reading
WATT Chron

POOR ORIGINAL

OFFICE ▶	DSS:RSB	DSS:RSB	DSS:RSB	DSS:RSB		
SURNAME ▶	JWatt:db	SIsrael	TNovak	DFRoss		
DATE ▶	2/2/78	2/2/78	2/3/78	2/3/78		

5.2.2 Overpressure Protection During Startup and Shutdown

Several instances of reactor vessel overpressurization have occurred in pressurized water reactors in which the Technical Specifications implementing Appendix G to 10 CFR 50 have been exceeded. Vessel stress limits as a function of pressure and vessel temperature decrease as the result of vessel irradiation through the life of the plant.

During the first fuel cycle, the applicant has administrative procedures and equipment to minimize the potential for excessive pressure transients under startup and shutdown conditions. By procedure, either a steam or nitrogen bubble will be in the pressurizer with a high level alarm and a low level interlock to maintain specified level limits. The presence of a bubble reduces the repressurization rate which results in more time for operator action. A single dual range relief valve will also be available.

The NRC staff has performed an evaluation of the Three Mile Island Unit No. 2 pressure vessel and determined that due to the small effects of radiation during the first fuel cycle, the allowable stress limits are not reduced below stresses resulting from overpressure events limited by safety valve set points with the vessel at ambient temperature. This evaluation provides the principal basis for concluding that an overpressurization event during the first fuel cycle would not present an undue hazard relative to vessel failure.

The applicant has provided a plant redesign incorporating a dual range setpoint for the pressurizer relief valve. During cooldown from hot shutdown, the NDTT mode for operation of the relief valve would be selected when the reactor fluid temperature is cooled to 275°F and the primary coolant pressure is below 450 psig. When in this mode, the relief valve would open should the pressure exceed 500 psig and the primary coolant temperatures remain below 275°F.

The applicant has evaluated this system considering seven different events representing the thirty events experienced in various PWRs. The analyses were performed with code DYSID. Credit was taken for administrative procedures requiring either a steam or nitrogen bubble in the pressurizer at all times. Credit was also taken for the pressurizer high level alarm and low level interlock to maintain the water between specified level limits. The results of the analyses indicated that reactor system pressure would not exceed 500 psig during any of the events.

The staff has reviewed the dual set point design and the results of the analyses to determine if adequate protection is provided through the life of the plant. The design does not meet the single failure criteria because only a single relief valve has been provided. Also, the code DYSID has not been reviewed by the staff.

The long term solution, which must be implemented prior to the second fuel cycle will require staff review and approval of the code DYSID and modifications to the present design to meet all of the requirements identified below.

1. Credit for operator action. No credit can be taken for operator action until 10 minutes after the operator is made aware that a transient is in progress.

2. Single failure criteria. The pressure protection system should be designed to protect the vessel, given any event initiating a pressure transient. Redundant or diverse pressure protection systems will be considered as meeting the single failure criteria.

3. Testability Provisions for periodic testing of the overpressure protection system(s) and components shall be provided. The program of tests and frequency or schedule thereof will be selected to assure functional capability when required.

4. Seismic design and IEEE 279 criteria. Ideally, the pressure protection system(s) should meet both seismic Category I and IEEE 279 criteria. The basic objective, however, is that the system(s) should not be vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

5. Reliability. The system(s) provided must not reduce the reliability of the emergency core cooling system or residual heat removal system.

7905010198

March 29, 1979

Harrell
Letter
1/10/79

Federal R ...
3/16/79 Exhibit 1157
B.N. on Non Safety Grade Equipment

MEMORANDUM FOR: Edward S. Christenbury, Chief Hearing Counsel

FROM: D. B. Vassallo, Assistant Director for Light W... Reactors, Division of Project Management

SUBJECT: BOARD NOTIFICATION - NONSAFETY-GRADE EQUIPMENT ... MITIGATE TRANSIENTS (BN-79-12)

The enclosed staff memorandum recommends that Boards be notifi... staff assessment to be made regarding the current practice of r... on nonsafety grade equipment to mitigate the severity of antici... operational occurrences.

As stated, there is no immediate safety significance to this is... although the assessment could lead to additional requirements o... equipment in the future. The timing of the Board Notification ... ation is based on the need for the staff to obtain information ... this matter from external sources.

Although the memorandum highlights the applicability to BWR's I ... that both BWR and PWR Boards in the appropriate time frame be m... of this matter.

POOR ORIGINAL

The staff memorandum should provide the basic Board Notification ... would suggest a covering paragraph along the following lines -

> "This notification (described in enclosed staff memorandum) i... the purpose of making the Boards aware of a staff assessment ... will be made regarding the current practice of placing relian... nonsafety grade equipment for the mitigation of the severity o... anticipated operational occurrences. It is of no immediate s... significance but could lead to a change in staff practice in ... future."

Our list of Boards currently in the notification time frame is a... follows:

POOR ORIGINAL

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

Lomic
Rosenthal

MAR 16 1979

MEMORANDUM FOR: D. B. Vassallo, Assistant Director for LWRs, DPM

FROM: R. L. Tedesco, Assistant Director for Reactor Safety, DSS

SUBJECT: BOARD NOTIFICATION - RECENT ISSUE ON NONSAFETY-GRADE EQUIPMENT (NSGE) TO MITIGATE TRANSIENTS

In analyzing anticipated operational occurrences (AOO), applicants (particularly BWRs) have relied upon normal operating equipment to mitigate the severity of the events. This equipment is not specifically qualified to seismic Category I or IEEE-279 requirements and has been termed nonsafety-grade equipment (NSGE); e.g., turbine bypass valves, relief valves, high water level trip, feedwater flow control, and pressure regulator. The reliability of such equipment has not been systematically evaluated by the staff and the issue focuses on the system design criteria that should be established for such equipment which is relied upon for mitigation, but to a lesser degree than required for other more severe events such as a LOCA or main steam line break.

NRC Office Letter No. 19 calls for a determination of the safety significance of new information by evaluating "whether this information could reasonably be regarded as putting a new or different light upon an issue before boards or as raising a new issue." Staff evaluation has reached a stage which concludes that this matter could be interpreted as raising a new issue. More information from a source or sources external to the staff is required to further study this issue. I therefore conclude that the notification test is met. In accordance with NRR Office Letter No. 19 requirement, I am providing you the following:

1. The item for notification

2. Considerations regarding relevancy and materiality

3. Statement on perceived significance

4. Relation to projects

Contact: T. M. Novak, NRR
49-27460

ORIGINAL

1. The Item

In the analysis of anticipated operational occurrences (AOO), credit has been given to non-safety grade equipment (NSGE) performing their normal function which can result in significant reductions with regard to the severity of the transient. The staff has not reviewed the design requirements for this equipment. Most of the equipment, however, is in normal plant usage in various control systems.

2. Relevancy and Materiality

The issue is relevant to evaluating the consequences of transients for all LWRs; however, it appears to be of particular concern for BWRs at this time.

3. Significance

In over 170 reactor years of normal BWR operations there has been no reported abnormal operational occurrence which has resulted in exceeding a technical specification safety limit. Based on this operating experience, we believe the real probability of an anticipated transient combined with an undesirable control system response which would result in violation of fuel damage criteria is low.

However, General Design Criterion 29 states:

> "Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences."

In addition, 10 CFR 50.55a, "Codes and Standards," Section (h) requires that protection systems meet the appropriate edition of IEEE Std. 279. These criteria have been applied to the primary reactivity control systems, e.g., the reactor scram system and to safety systems, e.g., ECCS.

While operating experience indicates that there is no immediate safety significance to this issue, the General Design Criteria suggest that additional reviews to ensure adequate thermal margins is warranted. Such margins could come in the form of additional equipment surveillance requirements, equipment modifications, or reanalyses of certain anticipated transients without taking credit for nonsafety-grade equipment (thereby affecting operating limits).

4. Relation to Projects

This relates primarily to BWRS. Since we view this issue as not having an immediate safety significance, operating reactors would be expected to await the outcome of our future reviews.

Appropriate boards should be aware of this issue and of the staff's plans to continue to evaluate its significance with regard to the acceptability of BWR transient analyses. It is anticipated that initial decisions on the direction of resolution on this issue would be available in 1979.

R L Tedesco

Robert L. Tedesco, Assistant Director
for Reactor Safety
Division of Systems Safety

cc: R. Mattson
V. Stello
T. Novak
R. Satterfield
V. Benaroya
G. Mazetis
S. Israel
R. Frahm
W. Mills
C. Graves
S. Hanauer

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

MAY 20 1977

Exhibit 1158

Note to: D. Eisenhut

From: D. Ross, Jr.

I am writing to express my concern over a growing trend by DOR:OT to develop capability redundant to DSS in several technical areas. More detail is provided in Enclosure 1, entitled "On Centers of Technical Excellence."

The conclusion drawn by me is that DOR growth in these areas has been in the name of high priority to operating reactors, yet many of the activities have little or no connection to the two traditional DOR missions; i.e., operating problems requiring immediate NRR safety decisions, and review of reload applications.

With NRR staffing at what appears to be a near-maximum, I believe that redundancy, not always an ideal factor in any case, to be a near-frivolity. The two technical work categories that I refer to in general are review of topical reports on methods, and development of solutions to generic problems. In some areas, as described in Enclosure 1, we are replicated to a needless degree.

Further, I am concerned with the lack of assurance of uniform technical output.

I recommend that certain technologies, including fuels, physics, and thermal-hydraulic methods, not be replicated. Within NRR we should have only one center of excellence for these. It may be a fair comment that in other disciplines there is also needless replication, but I am not in possession of any information to that point.

Enclosure 2 is a collection of the functional responsibilities of the four divisions. I do not know if it represents "official" policy. It says DSS should develop improved methodology and bases for carrying out reactor safety reviews, new design methods, etc. I do not see the phrase about "new design methods" in the DOR table, although that may not be particularly relevant.

Also enclosed is Table 1, a listing of some efforts in OT:RSB which seem to bear a lot on methods, computer calculations, basic research, etc. which I contend belong in a technology center.

Let me know your reaction.

D. F. Ross, Jr.

Enclosures:
As stated
cc: R. Heineman, V. Stello, E. Case

ENCLOSURE 1

ON CENTERS OF

TECHNICAL EXCELLENCE

Every large engineering organization eventually finds it necessary to establish centers of technical excellence within it. These centers are both sources and sinks of new methodologies. New methods and processes are assigned to the centers for review. Also, original processes and ideas are formulated by the centers and issued to operating groups.

These centers operate under different names; it may be long-range planning or research and development, or in our case (as I prefer to believe) reactor safety. At present, both the CPB and AB are intended to be centers of excellence in the areas of fuels, physics, and transient and accident analysis. In each discipline we have expert phenomenologists and expert analysts who on the one hand work to understand and interpret physical reality and on the other formulate and evaluate complex computer simulations (models) of the phenomena. These groups oversee large contracts with other laboratories to facilitate the review, analysis, and development of new and improved methods.

Now it seems the first fleecy clouds of an internecine storm are gathering. Within DOR there are people engaged in running the fuel performance code GAPCON, and developing commentary on its constituent parts. Within DOR there are people wishing to run the various physics codes such as PDQ and MEKIN. Within DOR there are people wishing to run the ECCS EM that AB is now working on, and the soon-to-be-developed IRT code for system transients.

This trend is to be deplored because:

1. It represents a drift towards replication of centers of excellence which are not needed in an organization of the limited size of NRR; it is therefore wasteful.
2. The work product of engineers and physicists in a center is most useful when it is compared with other efforts in the same technology and when reviewed at the same management level. The separation (as it is now developing) is producing conflicting judgments.
3. Many of the complex codes are sufficiently difficult that even a bright and resourceful engineer who happens to be organizationally separated from the development center may well go awry in his execution on interpretation. A symbiosis effect is necessary.
4. The redundancy is reflected at counterpart technical assistance laboratories, and is creating confusion and divisiveness there, also.

Briefly examined below is the DOR involvement in two traditional technical review areas; physics and fuels.

<u>Physics</u>

The current staffing of the Reactor Safety Branch of DOR includes nine physicists. (A tenth resides in the Plant System Branch.) Two of these (Chatterton and Sheaks) are still performing tasks that they began as members of the Reactor Physics Section of CPB and which remain the responsibility of the CPB.

Beyond evaluation of operating problems encountered at a specific plant and the collection and evaluation of operating reactor physics data, the physics responsibility of DOR entails the physics review of reloads which is generally fairly routine. (Occasionally, a different fuel vendor supplies a reload and has methods, principally related to power distribution control, which must be checked, but historically CPB has done this and can continue to do so.

There are signs lately that there are too many physicists for these DOR tasks. (A recent meeting on Exxon power distribution control was attended by seven DOR physicists and one DSS physicist.) This had led to talk of these people being used to run reactor physics codes such as PDQ-07, ARMP, XTG, and MEKIN to study various (generic) physics problems. It has also been indicated that some of these people would help "manage" the BNL physics program. (Two-thirds of the funds are provided by DSS and the DSS programs are effectively managed!)

Fuels

Riggs, Rubin, Lobel, Coffman and Baer work on rod bow. Since August 1976, DOR has written nine memos dealing with generic aspects of rod bow or meetings with fuel vendors. While some overlap is desirable, most flagrant intrusion involves Rigg's two lengthy memos describing the model he has developed for bow magnitudes.

Mendonca, Rubin, Coffman, Lobel and Baer work on fission gas. Baer, with Coffman's help, wrote a five-page memo criticizing DSS actions and recommending no interim or final licensing action. Mendonca, in a seven-page memo concluded that the effect was real and that the DSS correction was generally conservative and suggested some mechanistic inaccuracies in the DSS correction. Mendonca and Rubin ran a number of GAPCON runs trying to simulate the Zorita, AERE, KRB and other data sets. They also did GAPCON parametric studies. Total computer output about 2 ft. high, and effort is described as "quite extensive."

FY'77 includes a DOR technical assistance program to study "the effects of fuel rod volatiles on stored energy and fuel rod internal pressure." This was originally funded at 15K; although 10K was added, the scope was enlarged to include iodine spiking, which may be legitimate DOR generic work. FRAP-T was evaluated for 20K; this task has been admitted by DOR (Lobel) to be DSS work, but we didn't have enough money and they had too much, so they did it. FRAP-T evaluation and use for several generic accidents is included in FY'78 plans for DOR. Amount is not specified, but DOR funding for fuels is (proposed) up by 20% (CPB fuels proposal is down 5%).

ENCLOSURE 2

FUNCTIONAL RESPONSIBILITIES OF THE DIVISION OF PROJECT MANAGEMENT

-- Coordinates reviews of reactor applications through the operating license stage.

-- Determines the acceptability of license applications for docketing and develops schedules for carrying out the reactor safety review process in cooperation with DSS, DSE, and OELD.

-- Coordinates and particiaptes in the reactor safety review process by conducting meetings and preparing SER's and related documents, with input and contributions from DSS and DSE.

-- Coordinates reactor safety reviews with applicants, ACRS and ASLB.

-- Develops and reviews reactor SER's for completeness and consistency and obtains, in cooperation with DSS and DSE, concurrence of OELD.

-- Carries out technical aspects of reactor safety reviews of QA, emergency planning, industrial security programs, and financial qualifications.

-- Identifies, evaluates, and recommends confirmatory research programs to Director, ONRR.

-- Issues, upon proper approval and authorization, reactor licenses and authorizations (i.e., LWA's, CP's, PDA's, FDA's, and OL's).

-- Coordinates safety review of proposed government reactors exempt from licensing, e.g., DOD, ERDA, etc., and other special project reviews.

-- Administers the Commission's operator licensing function in accordance with 10 CFR Part 55.

-- Coordinates and participates in the public hearing process for reactors.

-- Carries out project management and safety reviews of advanced reactors through the operating license stage.

-- Coordinates with DSS, DSE and DOR to assure continuing uniform application of policy and technical positions to licensing activities and to assure that any new positions are consistent with the needs of each Division.

-- Carry out such project management or other activities as may, from time to time, be assigned by the Director, ONRR.

Division of Systems Safety

- Reviews and evaluates applications for nuclear power reactors, including advanced reactors, through the operating license stage.

- Develops safety evaluations for all systems and components of the proposed plants in accordance with Standard Review Plans and other design criteria and recommends schedules for carrying out reactor safety reviews in cooperation with DPM, e.g., custom, standard, duplicate and replicate plant designs.

- Evaluates systems for their impact on accident analyses and radioactive effluents and coordinates environmental impacts of plant with DSSEA.

- Develops improved methodology and bases for carrying out reactor safety reviews, new design methods, features, and arrangements, including impact-value analyses, incorporates such changes into the reactor licensing process, and recommends changes in Standard Review Plans, Regulatory Guides, Regulations, and policy, in accordance with prescribed change procedures.

- Develops solutions to reactor plant safety or licensing problems, utilizing resources from other Divisions of NRC and technical assistance contracts to obtain specialized service; coordinates improvements in regulatory guides and standards with OSD; and participates in the development of technical specifications to improve safety of reactor plants.

- Evaluates significant safety questions arising from operating reactors for their generic impact on original design of nuclear plants.

- Coordinates with DOR on safety evaluations of significant safety issues that would affect operating reactor plants.

- Identifies, evaluates, and recommends confirmatory research to be performed by RES with concurrence of the Director, ONRR.

- Participates in public hearing process associated with all reactor applications through the operating license stage.

- Reviews safety of proposed government reactors exempt from licensing, e.g., DOD, ERDA, etc.

- Carries out such licensing review or other activities as may, from time to time, be assigned by the Director, ONRR.

January 16, 1976

FUNCTIONAL RESPONSIBILITIES OF THE DIVISION OF OPERATING REACTORS

— Amends operating reactor licenses upon satisfactory conclusion of safety and/or environmental reviews with assistance from DSS and DSE as needed.

— Reviews operating reactor experience to assure that new findings are identified and incorporated in later reactor safety reviews conducted by DSS and DSE, and that such experience is applied to other operating reactors, as appropriate.

— Reviews operating reactor problems to assure that safety and environmental requirements are being satisfied and to assure that such problems are corrected with due consideration for safety and environmental protection.

— Coordinates all activities between NRR and I&E.

— Develops improved methods and bases for carrying out reviews of operating reactors, including procedures for incorporating experience from reactor reviews conducted by DPM, DSS, and DSE, taking into account impact/value analyses, to assure conclusions concerning operating reactors are balanced and safety reviews of operating reactors include all pertinent experience.

— Reviews, as requested, the safety of operational and design modifications of operating government-owned reactors exempt from licensing, e.g., DOD, ERDA, etc.

— Carrys out such licensing review activities of operating reactors or other activities as may, from time to time, be assigned by the Director of NRR.

— Carrys out, including contracting for, technical activities in support of operating reactor reviews and for the solution of problems needed to improve the regulatory process.

— Identifies, evaluates, and recommends confirmatory research programs to the Director of NRR.

— Performs ongoing assessments of operating reactors to determine the degree of compliance with current licensing regulations and standards.

— Conducts evaluations and issues construction permits and operating licenses for non-power reactors.

— Requests technical assistance from DSS and DSE, as required, to aid in providing assessments and solutions to current problems in operating reactors.

— Reviews and assesses proposals for the decommissioning of operating reactor facilities and the termination of licenses.

— Reviews requests for and grants exceptions to Title 10, Chapter 1 of the Code of Federal Regulations.

— Develops and maintains standard technical specifications for reactor facilities and assists in the application to new operating licenses.

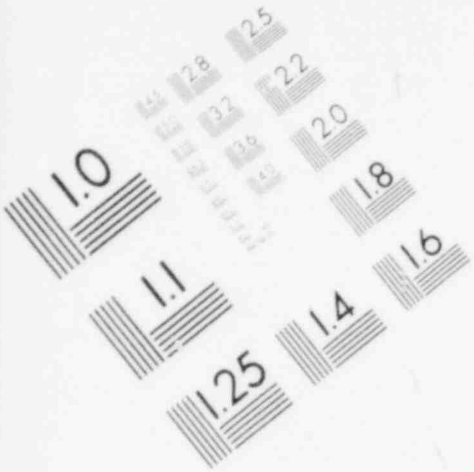
— Coordinates with DPM, DSS, and DSE to assure continuing uniform application of policy and technical positions to licensing activities and to assure that any new positions are consistent with the needs of each Division.

-- Audits OL reviews for power facilities and participates in the development of technical specification requirements prior to the transfer of responsibility for the project to DOR.

FUNCTIONAL RESPONSIBILITIES OF THE DIVISION OF SITE SAFETY & ENVIRONMENTAL ANALYSIS

- Reviews and evaluates site and environmental aspects of applications for nuclear power reactors, including advanced reactors, through the operating license stage.
- Performs project management and technical review functions of proposed sites where no specific CP license application is pending.
- Develops site safety and environmental analysis for proposed facilities in accordance with Standard Review Plans, NEPA, and design criteria and recommends schedules for carrying out environmental and safety reviews.
- Evaluates sites and systems for their impact on the analysis of accidental and normal release of radioactive effluents and evaluates environmental impacts of plants
- Develops improved methodology and bases for carrying out site safety and environmental reviews, new design methods, features, and arrangements, including impact-value analyses, incorporates such changes into the reactor licensing process, and recommends changes in Regulations, Regulatory Guides, Standard Review Plans, and policy, in accordance with prescribed change procedures.
- Develops solutions to site safety, environmental or other licensing problems, utilizing resources from other Divisions of NRC and technical assistance contracts to obtain specialized service; coordinates improvements in regulatory guides and standards with OSD; and participates in the development of technical specifications to improve safety of reactor plants.
- Evaluates significant safety and environmental questions arising from operating reactors for their generic impact on original design of nuclear plants.
- Coordinates with DOR on safety evaluations and environmental impacts of issues that would affect operating reactor plants.
- Identifies, evaluates, and recommends confirmatory research to be performed by RES with concurrence of the Director, ONRR.
- Participates in ACRS meetings and the public hearing process associated with all reactor applications through the operating license stage.
- Reviews safety of proposed government reactors exempt from licensing, e.g., DOD, ERDA, etc.

- Carries out such licensing review or other activities as may, from time to time, be assigned by the Director, ONRR.



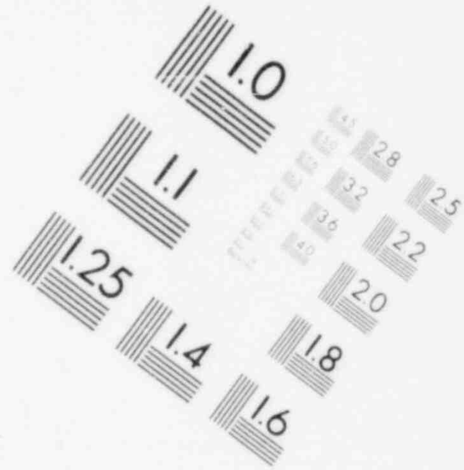
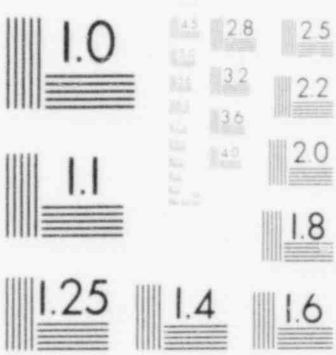
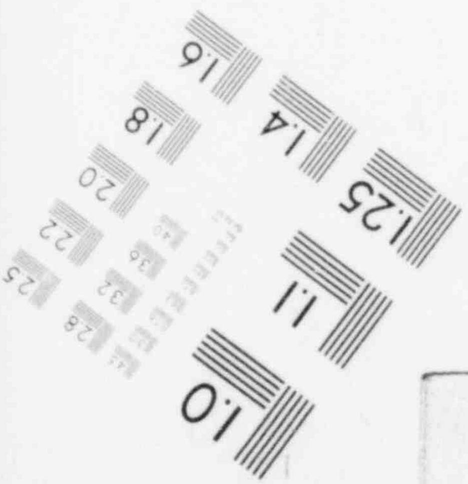


IMAGE EVALUATION
TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART



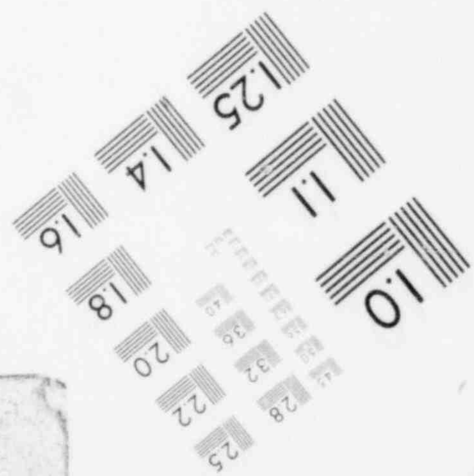


TABLE 1

TACS ITEMS OF INTEREST

AS OF 5/17/77

Lobel

LWR Fuel Behavior Research

Maine Yankee Analytical Models

Enhanced Fission Product Release for High Burnup Fuel

Rod Bowing

Anderson

ECCS Analysis Assistance

Chatterton

Reactor Startup Physics Tests

Coffman

Research Review Group - Zircaloy

Fission Gas Release

Mendonca

Fission Gas Release

Riggs

Rod Bow

Rubin

Rod Bow

VanderMolen

Noise Analysis - including Tech Assistance

Update of Cross-Section Files

Physics Effects of Perturbations in Fuel Geometry

Giannelli

Maine Yankee Analytical Models

Hardin

Maine Yankee Analytical Models

PCI Task Force

Landry

ECCS Analysis Assistance

PCI Task Force

Rosenthal

Deletion of APDMS at Beaver Valley

Sheaks

Review of INCA - CENPD 145

Excore Detector Response to Core Barrel Motion

Review of CENPD-153 - Uncertainties Related to Fixed In-Core

P_u Recycle

Tech Assistance - BNL Reactor Physics

Weiss

P_u Recycle

Noise Diagnostics

EXhibiT
- 1159

NOTE TO: D. Ross, Deputy Director, DPM DATE: ~~May 18~~, 1979

FROM: D. F. Bunch, Director, PSS

SUBJECT: TMI

D. Basdekas called to express a concern that inadequate attention was being given to reviews of the control systems of PWRs and their effect on plant thermo-hydraulic stability. He expressed the view that the staff had been directed not to review control systems in the past, and that such systems could interfer with the safety of the plants and that plants should be derated until satisfactory analyses have been performed. As I understand it, he believes that action taken in response to the TMI bulletins do not adequately address this area, nor does NUREG-0560.

Dil B.

D. F. Bunch, Director, PSS

cc: R. Tedesco

POOR ORIGINAL

50-320
Exhibit 1160 5.1
B&W

MAY 17 1979

MEMORANDUM FOR: H. Denton, Director, Nuclear Reactor Regulation

FROM: D. Ross, Jr., Deputy Director, Division of Project Management, NRR

SUBJECT: CONCERNS OF R. MCDERMOTT

References:
(1) Note from D. Ross to B. McDermott of 5/8/79
(2) Note: R. McDermott to D. Ross (thru W. Haass and D. Skovholt) of 5/14/79
(3) Memo W. Haass to D. Ross of 5/15/79
(4) Memo D. Skovholt to D. Ross of 5/15/79

The references (1-4) (copies enclosed) contain a dialogue on a contrasting technical viewpoint. I believe that the a-d items on p. 7 of Ref. 2 have been adequately treated in our ongoing work on Oconee. In due course I will ask Bob to review our SER and see if he shares this view. Bob's management has commented. W. Haass believes we should reexamine the 20-min HPI operation (as it might contribute to small LOCA probability) and the RCS pump operation.

D. Skovholt agrees with Walt's comments. These will be reexamined on 5/17.

This memo and the four references shall be placed in the PDR, in the docket file for each of the five B&W utilities under orders.

D. F. Ross, Jr., Deputy Director
Division of Project Management

Enclosures:
As stated

cc: T. Novak
V. Stello
W. Haass
D. Skovholt
R. Mattson
E. Case
R. McDermott
D. Crutchfield
R. Boyd
R. Reid
M. Fairtile

Docket Files (Oconee, ANO-1, Rancho Seco, Davis-Besse 1, Crystal River)
PDR
D. Ross Reading
NRR REading

DPM
DRoss:lk
5/17/79

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

May 8, 1979

NOTE TO: B. McDermott

FROM: D. Ross, Jr.

SUBJECT: SAFETY ASSESSMENT OF OPERATING B&W REACTORS

As you know, we are in the process of issuing confirmatory orders on all operating B&W-designed reactors. These orders have some conditions to be met as a basis for continued operation, both short-term and long-term. The bases for these orders may be found either in the April 25 NRR status report or in the generic B&W feedwater transient report (See R. Tedesco for a copy).

Through your efforts in evaluating the responses to bulletins issued to utilities using the B&W design, you have generated several notes and memos that relate to safety of continued or resumed operation of these plants. As you know, we are developing information on Oconee based on your inputs relative to shutdown or continued operation. We are also pursuing the same topics on other reactors. The topics include role and reliability of AFW; response to small breaks (including stuck-open PORV); natural circulation; and transients which have the tendency to produce primary voiding.

The items covered in your several notes are, in my opinion, being addressed by both the staff and the regulated industry. In all likelihood we will be developing soon the basis for continued or resumed operation for the B&W plants, for the short-term at least. However, as a double check, what I want you to do is to go over your memoranda, answer the following questions, and then give me your opinion, in writing, as to whether (a) your concerns are being addressed, and (b) whether the residual uncertainty is, in your opinion, too great to continue or resume operation.

Let me emphasize tha it is your expert opinion that is being solicited, so make this your individual effort. For the same reason it would not be especially useful to state "Michelson has a concern" (for example), unless you believe no one else is aware of your reference. In the interest of prompt resolution, try to have your report by Friday, May 11.

I re-read your memoranda to glean the topics; from them I got:

1. Performance of relief and safety valves under 2-ϕ or liquid conditions.

2. Performance of HPI pumps for small break at Davis Besse.

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3. Reliability of AFW.

4. Core cooling in natural circulation; with or without voids, with or without AFW, with or without consequential small break (PORV).

In order to be most useful to us you should use as an outline:

1. Statement of problem or concern.

2. Safety significance.

3. Your awareness of how it is being worked at NRC, and by whom.

4. Susceptibility to short-term resolution.

5. Your conclusion.

Let me emphasize that if there are other topics that I omitted, please fill them in. This work should reflect your own opinion, as ultimately we must decide whether this is a differing professional viewpoint in the sense of Office Letter No. 11. Also, since there are several topics, you should complete a summation to see if the cumulative effect is, in your opinion, so burdensome as to preclude operation of some or all B&W designed reactors even for a limited period.

Finally, you should re-read Office Letter No. 19 on Board Notification to determine whether you have any additional duties in that connection.

You should regard this assignment as your number one priority, as the decision on Oconee is but a few days hence.

Submit your report to me through W. Haass and D. Skovholt, who will each be asked to comment or concur.

D. F. Ross, Jr.

cc: E. Case
R. Boyd
D. Skovholt
W. Haass
R. Mattson
V. Stello
H. Denton

POOR ORIGINAL

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

MAY 14 1979

NOTE TO: Denwood F. Ross, Deputy Director, Division of Project Management

RJM for DJS see my comment to Ross dated 5-15

THRU: Donald J. Skovholt, Assistant Director for Quality Assurance & Operations, Division of Project Management

Walter P. Haass, Chief, Quality Assurance Branch, Division of Project Management

WPH (see comments to Ross memo dated 5-15-79)

FROM: Robert J. McDermott, Quality Assurance Branch, Division of Project Management

SUBJECT: SAFETY ASSESSMENT OF OPERATING B&W REACTORS

A. Introduction

This memorandum is in response to your May 8, 1979 note[1] to me regarding the safety assessment of B&W licensed reactors. Your note, I believe, was prompted by notes dated April 23, 24, and 25, 1979 that I forwarded to D. Eisenhut, DOR. This information was forwarded to D. Eisenhut based upon discussions I had with you regarding safety concerns for the continued operation of the licensed B&W reactors. As you know, it was at your directive that this information was channeled to D. Eisenhut, DOR.

In attempting to respond to your May 8, 1979 note, I feel it is appropriate and useful to offer some background information and my perspectives which follow:

1. Soon after the TMI-2 incident occurred (early April 1979) I was verbally informed that I was assigned to a task group which was chaired by Steve Varga, DPM, established for the purpose of evaluating licensee's responses to IE Bulletin 7905 (B&W licensed facilities). I participated on that task force for a period of approximately two weeks. During this time period, I and other members of the task group completed preliminary evaluations of the licensee's responses to Bulletin 7905 and a subsequent bulletin that was issued to holders of operating licenses for B&W reactors (7905A). It was during this period of time that, based on my individual review of the substance of the responses coupled with my personal knowledge of the technical aspects of the B&W reactors, my concern for the continued safe operation of B&W facilities began.

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[1] Note to R. McDermott from D. Ross

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2. My position relative to the continued use of nuclear power for both the short and long term, considering all alternate energy sources currently available, is that it is necessary. This position is predicated on the grounds that both the construction and operation of the facilities is conducted in a manner that provides reasonable assurance for the health and safety of the public from nuclear risks. It was in this spirit that I utilized resources available to me to promptly identify items which I considered to be of potential concern to NRR management. My actions were also prompted by my preliminary review of the bulletin responses, and my objective was to obtain information I believed relevant to reaching a prompt decision regarding the continued safe operation of licensed B&W reactors. My activity was accomplished in the background of numerous ongoing activities within the NRC staff relating to the TMI-2 event which I believed or perceived to represent an enormous burden on top NRC management and the staff in general. These numerous activities included special requests and inquiries from members of the press, Congress, Commissioners, the ACRS, staffing at the TMI site, support activities, etc.

3. My perception at the time the below listed information was being developed was that the B&W licensed reactors would be permitted to continue to operate (or restart and operate) for some extended period of time until management and staff resources could be made available using the existing organizational structure and available resources. Items which I considered to be important for immediate consideration included:

   a. A complete understanding of the design and operation of main and auxiliary feedwater systems for all B&W licensed reactors with the exception of TMI 1 & 2.

   b. Mechanistic ways which pressurizer code safeties or power operated relief valves could be actuated. My concern here was related to failure to reseat that could result in small breaks (steam or water side) that were below the lower bounds of the B&W generic loss of coolant accident analysis.

   Information supplied in the memos from R. McDermott to D. Eisenhut dated April 23 and 24, 1979 identified several potential problem areas with auxiliary feedwater systems at B&W reactors. Of particular note and concern was the fact that most of the reactors may not have enough installed auxiliary feedwater capacities (gpm) to satisfy the assumptions used in the B&W generic analysis[2] for small break loss of coolant accidents (i.e., B&W assumes 500 gpm per steam generator with auxiliary feedwater flow to each steam generator in the small break loss of coolant analysis). Additionally, my initial review of the information obtained from the licensees relating to the auxiliary feedwater systems was that in several instances for demand events

[2] Assuming single active failures.

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requiring auxiliary feedwater,* operator action would be required to initiate auxiliary feedwater flow. There also existed concern for the auxiliary feedwater systems at all facilities except Davis Besse 1 regarding the interconnection of the auxiliary feedwater system with the integrated control system whereby injection of auxiliary feedwater into the steam generator would be prevented by malfunctions or failures occurring within the integrated control system.

My initial review of information relating to mechanistic ways in which pressurizer code safety valves or power operated relief valves could open (i.e., system pressure reaching valve set points) disclosed that there were several plant transients initiated by malfunctions or failures in the secondary or balance of plant portion of the facility that would result in lifting of PORV or code safeties. Additionally, directives included in IE Bulletin 7905A imposed requirements for the plant operator to establish procedures to assure continued operation of the high pressure coolant injection pumps for a 20-minute period. This latter fact, coupled with the fact that there are several plant transients initiated by secondary system malfunctions that would automatically start Hpsi pumps and my perception that some operators would explicitly follow the prescription as outlined in the bulletin, heightened my concern because I was convinced that pressurizer code safety valves or PORV's would open in considerably less time than 20 minutes. An added concern is that available information obtained from the valve manufacturers for the code safeties and relief valves was that they stressed that the valves were only designed for steam service and that the effects on the valves from passing 2-phase or solid water through the valves were not known. The above stated concerns are related again to the possibility of creating small steam side or water side breaks that are smaller than those addressed in B&W generic analysis.

B. Summary of Information Provided to Date on B&W Reactors

Several items of potential concern were contained in enclosures to my notes to D. Eisenhut dated April 23, 24, and 25, 1979. A summary of each item is provided below.

April 23, 1979 note to D. Eisenhut - Subject: Information Applicable to B&W Reactors. The enclosure contained seven items of potential concern as listed below.

Item 1 - Summary of Possible Common Mode Failures of Auxiliary Feedwater Systems Observed in Operating PWRs. (4/22/79)

This issue was highlighted for management attention because there have been at least seven reported events where common mode failures have been reported to the NRC. It is my personal opinion that the common mode failures of auxiliary feedwater systems

*(assuming single failure)

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that have not been reported to the Commission would be many times the reported number. This is due in part, I believe, to the current wording of the technical specifications relating to LCO's (for the auxiliary feedwater systems) and the wording relating to the reporting requirements contained in the technical specifications. It should be noted that six of the seven common mode failures that have been reported to date resulted from system interactions with systems that are normally considered to be non-safety grade.

Item 2 - Available Information of Pressurizer Safety Valves for B&W Licensed Reactors. (4/23/79)

This information was provided to management primarily for the reason of identifying break area size that would result if a pressurizer safety valve fully opened and failed to reclose. In all plants reviewed (Crystal River, Arkansas 1, Rancho Seco, Oconee 1-3, and Davis-Besse 1) the equivalent break size area for a stuck open safety valve would be significantly less than the smallest break assumed in the B&W generic loss of coolant analysis. B&W's smallest break assumption is .05 ft^2. This was and is of concern because I believe the break size at TMI-2 was also significantly less than .05 ft^2.

Item 3 - Comments on Auxiliary Feedwater System Capacities. (4/22/79)

This information was provided for management's attention to highlight that four of the five plants reviewed (Crystal River 3, Rancho Seco, Arkansas 1, and Davis-Besse) may not have the capacity (gpm) equivalent to that assumed in the B&W generic loss of coolant accident analysis.[3]

Item 4 - Summary of Available Information on Pressurizer Code Safeties. (4/22/79

This information was provided to communicate my findings relating to information obtained from the manufacturers of the pressurizer code safety valves and their concerns related to 2-phase flow or solid phase flow through the code safeties. Davis-Besse 1 code safeties were supplied by Crosby Company and Dresser supplied safety valves for Oconee, Rancho Seco, Crystal River and ANO-1. Crosby representatives have stated in our communications with them that they believe that their valves will sustain damage on mixed (2-phase) flow or solid water flow. Both valve manufacturers stressed that valves are only designed for steam service and that they believe some damage may result from 2-phase or solid water flow through the valves. This item is of potential concern if Hpsi pumps are operated for a 20-minute period when 2-phase or solid water flow may be passing through these valves.

[3] If a single active failure in the auxiliary feedwater system is assumed.

Item 5 - Time for HPI to Lift Pressurizer Code Safeties. (4/23/79)

This information was provided to indicate if the bulletin directive (7905A) was followed explicitly (i.e., if Hpsi pumps started automatically, the operators should allow continued operation for a minimum of 20 minutes), it would be likely or probable that the pressurizer code safeties would lift in significantly less time than 20 minutes and that the possibility exists for mixed (2-phase) or solid water flow through the valves. See Item 4 above for potential safety concerns.

Items 6 and 7 - Comments on Oconee Feedwater Systems and Rancho Seco Feedwater Systems. (4/22/79)

This information was provided to identify potential areas of concern relating to the reliability of feedwater systems at Oconee and Rancho Seco. The summary provided in each of these documents highlighted several areas of potential concern relating to the reliability of these systems. This information was also provided to assure that management was aware of the assumptions relating the auxiliary feedwater flow rates that were utilized by B&W in their generic LOCA analysis.

April 24, 1979 note to D. Eisenhut - Subject: Additional Information Applicable to B&W Reactors. The enclosure contained three items of potential concern as listed below.

Item 1 - Comments on Arkansas Unit No. 1 Feedwater Systems. (4/24/79)

This information was provided to identify potential areas of concern relating to the reliability of feedwater systems at Arkansas Unit No. 1. The summary provided in the document highlighted several areas of potential concern relating to the reliability of this system.

Item 2 - Comments on Davis-Besse Unit No. 1 Feedwater Systems. (4/24/79)

This information was provided to identify potential areas of concern relating to the reliability of feedwater systems at Davis-Besse Unit No. 1. The summary provided in the document highlighted several areas of potential concern relating to the reliability of this system.

Item 3 - Comments on Crystal River Feedwater Systems. (4/23/79)

This information was provided to identify potential areas of concern relating to the reliability of feedwater systems at Crystal River. The summary provided in the document highlighted several areas of concern relating to the reliability of this system.

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April 24, 1979 note to D. Eisenhut - Subject: Additional Information Applicable to B&W Reactors. The enclosure contained one item of potential concern as identified below.

Item 1 - Calculation of Time For Makeup Pumps to Lift Pressurizer Safeties for Davis-Besse. (4/24/79)

This information was provided because of the unique aspects of the Davis-Besse 1 high pressure coolant injection system designed to mitigate small breaks in the reactor coolant system. Based on my review of the Hpsi system design of this facility (i.e., low head Hpsi pumps ~ 1650 psig shutoff head), the only mechanistic way to open to code safeties would be to operate the makeup pumps (these are not Hpsi pumps but the normal makeup pumps with a high shutoff head of 2774 psig.).

April 25, 1979 note to D. Eisenhut - Subject: Additional Information Applicable to B&W Reactors. The enclosure contained two items of potential concern as listed below.

Item 1 - Effective Break Size Calculations for TMI-2. (4/25/79)

This information was provided to assure that management was informed that the best estimate break size area for the TMI-2 event was 0.00729 ft^2, which is below the range of break size analyzed by B&W.

Item 2 - Comparison of Davis-Besse 1 ECCS to Other B&W Licensed Plants. (4/25/79)

This information was provided to assure that management was informed of the unique characteristics of the Davis-Besse 1 ECCS design.

C. Conclusions

Your note to me dated May 8, 1979 regarding the safety assessment of B&W operating reactors requested that I answer the following two questions and to identify any other areas of potential concern I had:

A. Whether my concerns are being addressed by the staff, and

B. Whether the residual uncertainty is, in my opinion, too great to resume operation.

In response to Item A, I believe that the staff is reviewing all of the items of potential concern identified to date by me in my notes to D. Eisenhut dated April 23, 24, and 25, 1979. That said, however, I have no current knowledge of the status of all the conclusions reached by the staff regarding my items of potential concern.

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In regards to Item B above, my personal opinions are as follows:

Currently licensed B&W facilities should be shutdown or remain shutdown until,

a. An analysis for each facility has been submitted and reviewed by the staff that shows conclusively that the core can be adequately cooled by natural circulation. Analysis should include evaluation of natural circulation with only one coolant loop in service and confirmatory testing should be conducted.

b. An analysis for each facility has been submitted and reviewed by the staff of small break loss of coolant accidents that is consistent with the capabilities of the emergency core cooling and auxiliary feedwater systems (as-built).

c. Each licensee has proposed and the staff has reviewed design modifications that will substantially improve the reliability and automatic availability of auxiliary feedwater systems above the existing levels. This item is not applicable to Davis-Besse 1.

d. Current directives which have been issued by the staff have been reviewed to assess the licensees' perception of the directives, the procedural implementation of the directives, and the anticipated operator response to anticipated operational occurrences in response to these procedures. The technical basis for the actions required by the directives, particularly, continued running of Hpsi and reactor coolant system pumps, should be provided to the owners-operators of the B&W facilities.

In summary, I believe that the above items are those that are, in my opinion, necessary and sufficient to provide reasonable assurance for the protection of the health and safety of the public from nuclear risk for the short-term (~4 months) until longer term corrective actions can be taken. This conclusion is based on my current understanding of B&W power plant designs that I reviewed and my concern that the continued use of nuclear power as a national energy source may be precluded if another TMI-2 incident were to occur.

Robert J. McDermott

Robert J. McDermott
Quality Assurance Branch
Division of Project Management

cc: H. Denton
E. Case
R. Boyd
R. Mattson
V. Stello
D. Skovholt
W. Haass

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

MAY 1 5 1979

MEMORANDUM FOR: Denwood F. Ross, Deputy Director
Division of Project Management

FROM: Walter P. Haass, Chief, Quality Assurance Branch,
Division of Project Management

SUBJECT: COMMENTS ON R. MCDERMOTT'S RESPONSE TO YOUR NOTE OF MAY 8, 1979

I have reviewed Bob McDermott's response of May 14, 1979 to your note of May 8, 1979, as summarized in the conclusions on page 7, and have the following comments:

1. Generally, I do not agree that the items, identified in the conclusions for completion prior to restart of the B&W plants, need to be accomplished within that time frame as qualified in comments 2 and 3 below. I do find one exception as noted in comment 4. I believe that, while the items identified should be considered in the overall assessment of the adequacy of the B&W plants with regard to safety, they are more appropriate for consideration under the long-term program as defined on page 7 of the Commission's Order to Duke Power Company. My rationale for this belief is that the staff has already developed, in my view, an acceptable program for the short-term, as described in the Commission Order (Section IV) and as required in the bulletins, that addresses the corrections necessary to provide the assurance that secondary system events are highly unlikely to result in a repeat of the TMI-2 accident. I believe the successful completion of the short-term program to be sufficient to permit restart of the B&W plants. The long-term program appears to be a satisfactory approach to treating related problem areas that are important but of a less significant nature.

   I note, however, that it is not obvious to me that the items identified in the McDermott response are included in the long-term program. This needs to be reviewed by the appropriate technical personnel.

2. Item b in the conclusions of the McDermott response appears to be largely similar to item (1)(d) (Section IV, page 11) of the Commission Order. This needs to be reviewed by the appropriate technical personnel.

3. Item c in the conclusions of the McDermott response appears to be largely similar to, or at least duplicative of to some extent, item (1) (including all its parts except d) of the Commission Order (Section IV, pages 9-12). This needs to be reviewed by the appropriate technical personnel.

4. Item 4 in the conclusions of the McDermott response expresses concern about continued operation of the HPI pumps for 20 minutes following reactor trip.

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Calculations performed by Don Beckham (Note to D. Eisenhut from R. McDermott, item 5, dated 4/23/79) indicate that the code safety valve will lift under these conditions. Speculation is that the valve will then pass water or a water/steam mixture potentially causing damage to the valve to the extent it may not reseat when system pressure drops. This may have happened at TMI-2 effectively resulting in a small break loss-of-coolant accident. Therefore, I believe that the technical basis for the staff-directed 20 minute HPI pump operation needs to be re-evaluated by our technical personnel prior to restart.

A similar re-evaluation may be necessary of the staff requirement for continued RCS pump operation without any apparent restriction. I am concerned that such an operating directive could possibly exacerbate an already poor situation.

Based on the above comments, I have not concurred in Bob McDermott's response to your note. However, as noted in several places above, the concerns expressed, as well as the information developed, the calculations performed, and the evaluations that resulted, should be brought to the attention of the appropriate technical staff members for further consideration.

Walter P. Haass

Walter P. Haass, Chief
Quality Assurance Branch
Division of Project Management

cc: R. Boyd
D. Skovholt
R. McDermott

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

May 15, 1979

MEMORANDUM FOR: D. F. Ross, Jr., Deputy Director, Division of Project Management, NRR

FROM: Donald J. Skovholt, Assistant Director for Quality Assurance and Operations, DPM

SUBJECT: COMMENTS REGARDING R. MCDERMOTT MEMO

I have reviewed the memorandum from R. McDermott dated May 14, 1979 regarding the safety assessment of B&W reactors.

As noted by Mr. McDermott, the work by him that led to his concerns was performed during the period that he was assigned to a bulletin-review task force. I noted that he performed his work with a high degree of diligence but, since this assignment is not related to responsibilities of my office, I have not performed the in-depth technical review that would be necessary to corroborate each point. However, I do offer the following comments regarding the concerns and conclusions in Mr. McDermott's report.

1. I believe that these concerns reflect valid questions that have resulted from consideration of the TMI-2 incident and appropriate staff investigation is warranted to resolve them.

2. I believe that these concerns are known to, and are being addressed by, the appropriate staff organizational units responsible for their investigation. I note that Mr. McDermott also believes that they are being reviewed although he indicates that he does not have current knowledge of the status of all the conclusions reached. Likewise, I do not continually have, or need to have, current knowledge of developments in all staff review areas; however, I have no reason to question the ability of the assigned staff to perform these functions.

3. With regard to Mr. McDermott's opinions concerning the need to keep all B&W reactors shutdown until a number of analyses and modifications are performed, I do not find a convincing basis to support this action as

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necessary for the protection of public health and safety. It seems to me that the expressed concerns are already part of the Commission's action program on a time scale deemed appropriate. Certainly, Mr. McDermott's concerns should be provided to the assigned review groups to take into account.

I have also reviewed the comments of Mr. Haass on the memorandum and am in agreement with them.

Donald J. Skovholt

Donald J. Skovholt, Assistant Director
for Quality Assurance and Operations
Division of Project Management

cc: R. S. Boyd
W. P. Haass
R. J. McDermott

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Distribution
~~Central File~~
NRR Reading
~~F. Williams Reading~~

May 17, 1979

MEMORANDUM FOR: Darrell G. Eisenhut, Deputy Director, Division of Operating Reactors

THRU: Steven A. Varga, Chief, Light Water Reactors Branch No. 4, Division of Project Management

FROM: F. J. Williams, Jr., Technical Coordinator, Division of Project Management

SUBJECT: CONCERNS RELATED TO TMI-2 EVENT AND BULLETIN 79-05A - DON QUICK (IE: REGION II)

On May 2 and 3, 1979, Don Quick of Region II made a verbal presentation to the S. Varga review group and other staff members (attendance list enclosed). Based on his own experience and discussions with other IE personnel he provided his assessment of general problems with operations and maintenance and with his perception of problems with the B&W design. In many instances he cited what he felt were specific design deficiencies and in other areas he described broader problems (e.g., waste disposal area and control boards) applicable to many or all designs.

Enclosure 1 provides my writeup of the presentation. It has been reviewed by Don Quick and his corrections and additions have been incorporated.

No assessment has been made of the concerns expressed. Some have been presented in other forms. Most of them, in my opinion, represent long range consideration. Some of the cited design deficiencies could bear on the current considerations regarding the B&W operating plants.

ORIGINAL SIGNED BY

F. J. Williams, Jr., Technical Coordinator
Division of Project Management

Enclosures:
As stated

POOR ORIGINAL

cc: H. Denton
E. Case
R. Mattson
V. Stello
D. Ross
S. Hanauer

R. Tedesco
S. Varga
F. Williams
Attendance List

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OFFICE	TC:DPM	LWR#4:DPM			
SURNAME	FJWilliams/bm	SAVarga			
DATE	5/1/79	5/1/79			

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1979 - 625-624

ENCLOSURE 1

COMMENTS AND CONCERNS EXPRESSED BY DON QUICK

General

Operators - The current situation with regard to operator capabilities is not good. Operators had a keener sense of awareness 15-20 years ago. Parameters were closely monitored to keep things on track - there was more concern as to potential problems when things were not on track and actual plant status was better known. The basic philosophy developed over the years is that plants are designed so that they will return into some stable condition following a transient. Operators have been convinced of this philosophy through training approach and have become overconfident. Many operators cannot explain effects of jumpers, clearances etc. on system response. We have contributed to the operator problem by our approach of extensive QA, check lists. As a result they are further convinced that nothing can go wrong if procedures are followed. In reality many emergency procedures are inadequate. In addition, we provide edicts without bases and without participation on the part of the operators. All of the above have resulted in a complacent attitude which applies to maintenance as well as operators.

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In addition - the training has been deficient. It follows along the classic accident lines and does not prepare the operator for the unexpected.

Maintenance - The attitude problems discussed above for operators also apply to maintenance. There are many instances of equipment being out of service too long - both safety related and non-safety related components - both in violation of Tech. Specs. and because Tech. Specs. are too lenient. Examples of problems - (a) ISI references Section XI which gives 96 hours to interpret test results - this can be added to the 72 hours permissible out of service to give a total of a week -

(b) the Tech. Specs. give 72 hours and some utilities utilize the full 72 hours rather than going full out to return the equipment to service as soon as possible (overtime vs. dayshift consideration).

Improvement - No easy solution is seen for the operator problem. The following items were presented as tending to alleviate the pr lem.

- Improve training (depth and scope for operators and maintenance personnel).
- Attempt to obtain more operational input for control board and system design.
- Improve selection of parameters presented to operators and manner of displaying the information.
- Encourage utility management to attempt to improve such things as; morale, working conditions, attitude toward both operations and maintenance, communication, and plant status awareness.

Details of Concerns (B&W Plants)

Heat Sink - The OTSG causes rapid effects on parameters during transients or trips e.g., pressurizer level changes are magnified vs. W or CE design. There is concern that this plant was licensed with this size pressurizer since many transients have given us either OTSG or pressurizer in a nearly dry condition. It is difficult to return to a stable condition following a transient - particularly the loss of feedwater transient. Accident analysis should be reevaluated in light of knowledge gained from the TMI event.

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There is a lack of OTSG level trips - the design was apparently based on fast recovery following secondary transients, and the primary system design lacks the capability to handle these transients under certain conditions.

It is recommended that OTSG level trips, a larger pressurizer and upgraded auxiliary feedwater (AFW) systems in light of the low mass and unreliable heat sink (OTSG) be provided.

AFW Systems

As stated under Heat Sink, above, he recommends upgrading the existing AFW systems to meet current requirements. There are general problems with lack of redundant flow paths, and pumps and poor indications of system operability. Some specific problems were listed based on familiarity with the Crystal River (CR) plant. CR has one auto steam driven pump and one manual electric driven pump. Surveillance procedures on the electric pump negate auto start of steam pump - the electric pump cannot be carried on the diesels unless other safety related loads are shed - if the motor driven pump has replaced operation of the steam driven pump the steam driven pump will no longer start on auto if we lose the motor driven pump.

At Oconee there is only one AFW system per unit - manual valving is required to align cross ties.

At ANO-1 the design includes a recirculation valve which must open to prevent overpressurization of AFW system under no-flow conditions.

At CR a break anywhere in the 6" startup feed line causes loss of all emergency feedwater.

At CR the steam break isolation matrix closes all valves (including emergency) to isolate affected OTSG on low pressure. However, without appropriate check valves it appears that you could isolate both OTSG's thereby requiring manual operations to supply emergency feedwater after determining which OTSG had the break.

There is no annunciation on unavailability of pumps or valves not in position. There is no emergency flow indication. It is recommended that AFW systems receive same treatment with regard to status indications as ECCS.

Natural Ciruclation

Natural circulation capabilities as well as procedures and the need for actual testing should be reevaluated. We should not be relying on isolated tests.

The capability of going into natural circulation with voids in the system should be evaluated. He thinks that there is a high probability of void existence as a result of the low pressure for HPI actuation (close to saturation).

Review interface with ICS which programs a higher level with no RCP's to enhance natural circulation.

ICS Interface with FW

The ICS in not designed as a safety related system but appears to be controlling a safety related system. In some designs it has only a single power supply - in others it requires a manual changeover to an alternate power supply. Westinghouse includes trips on many of the functions covered by the ICS therefore they can get away with a non-safety grade design. The B&W ICS should be upgraded.

PORV

The Bulletin requirements in the area of lower scram settings, increased PORV lift pressures and reactor trip on turbine trip alleviate some of the concerns with the B&W PORV.

The B&W PORV type provides poor position indication (PI) to operator - only indication is energization of solenoid which is indirect at best. Several years ago it was determined that no PI was available and that loss of power caused the valve to fail open. This led to current PI but some PORV block valves do not have Class IE power supply. There is a history of valve failures and he is of the opinion that the valve should be redesigned. The B&W valve was never qualified to pass water and function properly.

The Westinghouse PORV is air operated with direct PI limit switches on valve travel.

HPI

Initiation point is too close to saturation. The capacity of HPI to remove decay heat (without heat sink) is questioned. The staff should relook at cooling capacity and reevaluate procedures which call for the operator to throttle flow. All procedures calling for operator to throttle a core cooling flow should be reviewed.

Level-Pressurizer & OTSG

Should look at B&W plants for trips based on OTSG level. Westinghouse has several trips - level - mismatch etc. B&W has no trip on pressurizer level. Westinghouse has a high level trip. There appears to be an inconsistency in the review of these designs.

Recommends the addition of trips for the B&W plants in addition to upgrading the level instrumentation to safety grade on both pressurizer and OTSG.

The FW trip is not always effective - in some cases the FW pump is set back to minimum speed. This results in no real feed flow but does not give reactor or turbine trip.

Reactor Coolant System Instrumentation

Existing instrumentation is not properly placed on RCS to give operator the information needed to assess his problem.

When bubble formed (TMI-2) pressure and temperature indicated subcooling, therefore, TH is not providing proper indication of the upper plenum temperature or there is a lag time.

Should have instrumentation in highest system points to evaluate natural circulation performance.

No plants - B&W, Westinghouse or Combustion Engineering (except for some older plants) have upper head temperatures.

The types of installed instruments must be reliable under voided conditions. Westinghouse Tave may be very inaccurate if correct bypass flow is not available.

Control Board

Not enough human engineering or operations input has gone into the design and layout of the control board.

Control boards have not been designed to permit effective operation under a transient condition.

Key parameters should be identified for voiding or LOCA situations. These parameters should be grouped in a display configuration so that they are readily apparent to the operator.

H_2 Recombiners

CR does not have a recombiner - they rely on purge which is totally inadequate.

As a result of TMI-2 we need to take another look at containments from the standpoint of dead ended volumes etc. in order to prevent local buildup of explosive mixtures.

Gaseous venting from the RCS has not been a concern in the past but obviously needs thought now. One possible design approach is the installation of vents from RCS high points to the pressurizer gas space with controlled bleed from that point.

Containment Isolation

All vendors have unique problems. Most B&W plants use only a containment pressure signal which is not satisfactory.

Bulletin calls for isolation of lines not essential to core cooling on Safety Injections. Thinks this is a vague requirement and doesn't recommend that we isolate RCP cooling water, cooling to rod drive motors or cooling to ventilation system coolers. We should keep them available until a containment pressure signal isolated them - don't isolate or safety injection - keep cooling water available to the containment.

Another potential isolation problem is applicable to St. Lucie 2 where HPI pumps don't inject at normal system pressure. Isolating normal charging path could then result in no injection at higher pressures.

Waste Disposal

The tankage and waste processing has been inadequate for years. Not enough attention has been paid to the waste disposal systems - in many cases these systems were field designed and installed. Have always given a lot of problems with regard to operator exposure and accessability. Most licensees have run borderline on waste disposal capability.

TMI-2 has only highlighted this existing problem.

One sump pump contaminated a whole building. Why does the design include a line from the containment building sump to the auxiliary building sump? This exists on several plants.

Controls are generally located in a part of the building with difficult access and small events have resulted in serious inaccessability.

Bulletin Comments Based on Plant Visits

What is interpretation of running RCP? Does it mean run to destruction? Is vendor sure the pump can operate in a steam environment? Obviously need to consider requirement to have pumps available during recovery phase of any accident.

Why run HPI for 20 minutes? The subcooling criterion appears to be the important concern.

General concern - unless the operators are aware of our bases for some of these edicts we can't expect to get their cooperation.

ENCLOSURE 2

ATTENDEES - DON QUICK PRESENTATION

MAY 2, 1979

NRC

R. McDermott
A. Oxfurth
B. Clayton
B. Boger
N. Wagner
T. Wambach
M. Williams
H. Silver
Frank Orr
R. W. Woodruff
Harry Rood
Don Quick

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

APR 10 1979

Exhibit 1162
no copies

MEMORANDUM FOR: The Files

FROM: R. L. Tedesco

SUBJECT: WESTINGHOUSE ACTION ON TMI-2 INCIDENT

In a telecon held on April 10, 1979, Tom Anderson described various actions that were being recommended to the owners of W PWRs concerning the TMI-2 incident. Included was one of particular significance that deals with SI resulting from coincident signals of Low Pressurizer level and pressure. I have enclosed a copy of the W statement that indicates that SI may need to be manually initiated for TMI-2 type of events because of uncertain level indication in the pressurizer. This action is applicable to the operating W-PWR.

R. L. Tedesco

Enclosure:
As stated

cc: E. G. Case
F. Schroeder
D. Eisenhut
P. Check
G. Lainas
N. Moseley (I&E)
G. Mazetis
D. Davis
D. F. Ross

POOR ORIGINAL

7907020127

DATE: April 10, 1979

TO: Robert L. Tedesco
Assistant Director for Plant Systems
Division of Systems Safety
U.S. Nuclear Regulatory Commission

FROM: Thomas M. Anderson, Manager
Nuclear Safety Department
Nuclear Technology Division
Westinghouse Electric Corporation

SUBJECT: Telecon of April 10, 1979

As discussed with you on the telephone this morning, I am attaching a copy of the Saturday, April 7, 1979, telephone notification which was made to all Westinghouse operating plant customers having coincident pressurizer pressure and pressurizer level safety injection initiation.

Written followup notification is in the process of being carried out. Similar written notification will also be provided, as applicable, to those utilities having plants under construction.

APPROVED: T. M. Anderson

T. M. Anderson, Manager
Nuclear Safety Department

POOR ORIGINAL

7907020130

stated to your representatives in our meetings of April 5, 1979 to review the implications of the Three Mile Island incident, you should insure that your operators are immediately provided with additional information and instructions not to rely upon pressurizer level alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions in the event a transient occurs. Your plant requires coincident low pressurizer pressure (P_p) and low pressurizer level (L_p) in order to actuate safety injection. Preliminary analyses of a small break in the pressurizer indicates that L_p may hang up while P_p continues to decrease. Westinghouse strongly recommends immediate actions to instruct your operators that P_p should be monitored carefully along with other important information. In particular, if P_p drops below the safety injection initiation setpoint for your plant, safety injection should be manually initiated. As additional information becomes available, we will communicate further with you regarding this matter. We ask that you keep W informed regarding changes that you deem appropriate as a result of your review. This is consistent with the recommendation made at our Thursday, April 5, 1979 meeting in Monroeville and is being reiterated here to insure that appropriate action is taken quickly.

a change of good [illegible] ... should

Technically

– [illegible] W [illegible], [illegible] W [illegible]

– [illegible] W recommends re securing SI

& dealing with uncertain level information

POOR ORIGINAL

(1) Parler

(2) Hebdon

(3) RDY

Exhibit 1163

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

JUL 27 1979

MEMORANDUM FOR: E. Case, Deputy Director, Office of Nuclear Reactor Regulation

FROM: D. Ross, Jr., Deputy Director, Division of Project Management

SUBJECT: MEMO. DEYOUNG TO DENTON "PRECURSOR EVENT IN A FOREIGN REACTOR" DATED 7/24/79

The subject memo (enclosed) asks four questions plus asks for all relevant information currently available. I have made the following arrangements:

a. I will answer questions one and two; most of the discussion will be contained in our generic W report. I will send the foreign report to DeYoung.

b. Moseley, IE, will handle the part 21 question, and answer question 3.

c. Faulkner, IP, will answer question 4.

The subject memo does not accurately represent W plants as currently configured. Paragraph 3 of Bulletin 79-06A (April 13, 1979) required that the low pressurizer level logic be kept in the tripped condition. No one else used coincidence logic.

We are in the process of distributing the report to the industry; see the Enclosure from H. Faulkner.

D F Ross

D. F. Ross, Jr., Deputy Director
Division of Project Management

Enclosures:
As stated

cc: R. DeYoung
T. Novak
N. Moseley
H. Faulkner

POOR ORIGINAL

Denny, Tom, Roger

I believe this memo is a result of my discussions with Hebdon. I have told Hebdon my understanding of the event, its impact on safety and who knew what & when. Ashd

July 24, 1979

In Reply Refer to:
NTFTM 790724-02

MEMORANDUM FOR: Harold R. Denton, Director, Office of Nuclear Reactor Regulation

FROM: Richard DeYoung, Deputy Staff Director
NRC/TMI Special Inquiry Group

SUBJECT: PRECURSOR EVENT IN A FOREIGN REACTOR

We understand that in 1974 a small LOCA occurred at a foreign reactor that is very similar to the TMI incident. During the course of the incident steam formed in the RCS hot leg causing pressurizer level to rise while RCS pressure continued to decrease. This void formation caused pressurizer level to increase despite the fact that primary coolant was still being released from the system. The protective system in this design, which is similar to many U.S. reactors, required low pressurizer level and low RCS pressure for safety injection to be automatically initiated. This combination of coincident initiating signals and increasing pressurizer level caused the failure of safety injection to initiate while a small LOCA was occurring. Since many U.S. reactors have the same coincident logic for initiating safety injection, they are susceptible to the same problem. In addition, if the ECCS system could be deceived by this transient and its effect on pressurizer level, then operators of plants with other designs could have been confused by the pressurizer level indication that resulted from this transient.

Despite the significance and relevance of this incident to U.S. reactors, to our knowledge this incident has never been reported to the NRC by the vendor involved. 10 CFR Part 21 and Section 206 of the Energy Reorganization Act of 1974 require the reporting of defects and noncompliances to the NRC. We understand that individuals subject to Part 21 need to report failures or defects in foreign reactors that could create a substantial safety hazard in facilities and activities in the United States. Based on the insights resulting from the TMI accident, it would appear that this incident should have been reported by the vendor following the TMI accident.

We request that all relevant information currently available to NRR concerning this event be forwarded to us as soon as possible. This information should include as a minimum:

OFFICE ▶						
SURNAME ▶						
DATE ▶						

1. A description of who within the NRC became aware of this event, by what means was knowledge of this event formally or informally received by the NRC, and when was knowledge of the event acquired.

2. A discussion of the basis for any decisions that have been made concerning the safety significance of this event and its applicability to domestic reactors.

3. A discussion of the regulatory requirements associated with the reporting of this event to the NRC by the vendor both after and prior to the TMI accident.

4. A discussion of the basis for any decisions to release to the public information associated with this event.

We request that we be kept informed of the status and eventual resolution of this matter.

/s/

Richard DeYoung
Deputy Staff Director
NRC/TMI Special Inquiry Group

DISTRIBUTION
TERA
RDeYoung
KCornell
GFrampton
PNorry
WParler
FHebdon
✓AThadani
FFolsom

OFFICE ▶	NRC/TMI	NRC/TMI	NRC/TMI	NRC/TMI	NRC/TMI	NRC/TMI
SURNAME ▶	FHebdon:mc	WParler	KCornell	GFrampton	PNorry	RDeYoung
DATE ▶	7/20/79	7/ /79	7/23/79	7/23/79	7/ /79	7/23/79

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

JUL 25 1979

MEMORANDUM FOR: D. F. Ross, Deputy Director
Division of Project Management, NRR

FROM: Howard J. Faulkner
Research Agreements Coordinator
Office of International Programs

SUBJECT: AGREEMENT OF CONFIDENTIALITY FOR FOREIGN REPORTS

NRC has received permission to pass the two reports, Technical Report on Besnau Unit One Incident of August 20, 1974: TG-1 Trip/Reactor Trip/ Safety Injection Actuation and Report on Special Event Number 74-13, Trip TG-1/Reactor Trip/SE (Safety Injection) Trip, to Westinghouse, EPRI, and NRC licensees and contractors in connection with their activities in light of the TMI-2 accident. Accordingly, a draft letter and agreement of confidentiality are attached for your use. Both of these documents have been reviewed and cleared by ELD.

I will forward the reports for EPRI directly to Mr. Edward Zebroski. You are authorized to transmit these documents to other appropriate parties in the above listed categories.

Please send the signed agreements of confidentiality to me for our records.

H. J. Faulkner
Howard J. Faulkner
Research Agreements Coordinator
Office of International Programs

Attachments:
1. Draft Letter
2. Agreement of Confidentiality

DRAFT
HJF:ecb
7/23/79

Dear Mr. _______________:

The attached foreign reports are provided to you in confidence for use in your reactor evaluations in light of the Three Mile Island Unit 2 accident. Please have an authorized official of your firm sign the attached agreement of confidentiality and return it to me. If, for any reason, your firm declines to sign the agreement of confidentiality, access to the information contained in these reports is not authorized to your firm. If this situation develops, return the foreign reports to me immediately.

If you should have any questions regarding this action, please contact Mr. Howard Faulkner at 301-492-7788.

Sincerely,

Denwood F. Ross
Deputy Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachment: 2

1. Technical Report on Besnau Unit One Incident of August 20, 1974: TG-1 Trip/Reactor Trip/Safety Injection Actuation
2. Report on Special Event Number 74-13, Trip TG-1/Reactor Trip/SE (Safety Injection) Trip

AGREEMENT OF CONFIDENTIALITY WITH THE NRC

The reports, Technical Report on Beznau Unit One Incident of August 20, 1974: TG-1 Trip/Reactor Trip/Safety Injection Actuation and Report on Special Event Number 74-13, Trip TG-1/Reactor Trip/SE (Safety Injection) Trip, are provided for the use of your organization in your review, evaluation, and assessment of nuclear reactors in light of the accident at Three Mile Island Unit 2. These reports are being provided under the following conditions:

1. The reports can be transferred from the NRC to appropriate NRC licensees, contractors, the Westinghouse Electric Corporation, Combustion Engineering and the Nuclear Safety Analysis Center of EPRI.

2. The reports will only be transferred, disseminated, disclosed or otherwise communicated, in whole or in part, to persons or organizations involved in the above task.

3. The information contained in the reports will not be used directly, indirectly, or otherwise, except as may be necessary to accomplish the task.

4. Information contained in the reports may be discussed and communicated between the recipients, but only in connection with the specified task.

5. The reports will not be duplicated or transferred in whole or in part, by the recipients.

6. The report will be destroyed at the completion of the task; this action will be certified in writing to the NRC.

7. All of the above conditions shall be made a part of any transfer permitted under (2) above.

____

Signature

____

Title

____

Firm or Organization