Transcript of Proceedings

UNITED STATES OF AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT
THREE MILE ISLAND

DEPOSITION OF: SAUL LEVINE

Bethesda, Maryland

August 8, 1979

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DEPOSITION OF: SAUL LEVINE

Room 6211
7735 Old Georgetown Rd.
Bethesda, Maryland

August 8, 1979
2:50 o'clock p.m.

APPEARANCES:

On Behalf of the Commission:

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E X H I B I T S

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1	Resume of Saul Levine	4
2	NRC Policy Statement on criteria for determining abnormal occurrences, published in Federal Register (42 FR 10950) on February 24, 1977, Abnormal Occurrences Reports"; and NUREG-0090, Vol 1, No. 4, Report to Congress on Abnormal Occurrences October-December 1978, published March 1979	9 & 33
3	Memorandum undated prepared approximately 8/31/76, from S. Levine, Nuclear Regulatory Research, and W.G. McDonald, Management Information Program Control, to Rusche, Nuclear Reactor Regulation; Volgenau, I&E; Minogue, SD; Chapman, NMSS; Subj: Development of Performance Evaluation Programs. (The Levine-McDonald Memo)	9, 33, 53
4	Letter from R. Salvatori, Manager, Nuclear Safety Department, Westinghouse, dated November 1, 1974 to Saul Levine, AEC, commenting on draft copy of WASH-1400	41 53
5	Memo to File, Nuclear Regulatory Commission dated January 19, 1977 written by Joe LaFleur re Gundremmingen incident	54
6	Document entitled "Abnormal Occurrence at the Nuclear Power Plant Gundremmingen" dated January 13, 1977 with attached report to the Bravarian Parliament, draft, dated January 19, 1977, informal translation	54
7	Department of State telegram from American Embassy Bonn, Germany, to Secretary of State addressing the January 13, 1977 incident at Gundremmingen Nuclear Power Plant	54

P R O C E E D I N G S

WHEREUPON,

SAUL LEVINE

was duly sworn by Gary M. Sidell, Esquire, and was examined and testified as follows:

DIRECT EXAMINATION

BY MR. SIDELL:

Q Would you state your name for the record, please?

A Saul Levine.

Q And your current position with the NRC?

A Director of the Office of Nuclear Regulatory Research.

Q Were you requested to bring a resume or biographical description with you today?

A Yes, I was.

Q Have you brought that?

A Yes, I have.

Q May I have it, please?

A Yes.

Q Is the information as contained in your resume complete and accurate to the best of your knowledge?

A I don't know exactly what you mean by complete. It is essentially complete, yet. There are minor duties that are not spelled out in there, but it's complete and accurate.

Q It's complete in terms of the fact that it includes major areas and professional responsibilities you have?

A Yes, it does.

Q Let's mark this as Exhibit 1 to the Deposition.

(WHEREUPON, the document referred to was marked as Exhibit 1 to the Deposition.)

Q In your position as Director of the Office of Nuclear Regulatory Research, could you tell me what the distinction might be with the term regulatory research as opposed to an alternative kind of research?

A Yes. When people talk about research they generally think of two kinds of research. One would be pure research which is just basic science without application. Another is generally called R&D, research and development, where the result is a product, a space shuttle, or what have you. When the Congress established the NRC they coined a new term called confirmatory research to differentiate it from research and development of a kind normally done by the AEC.

Research and development has a promotional connotation. That is, someone has an idea for a project and people go out and sell the utility and need for that project and the great things it will do for you either in terms of convenience or leverage in the economic sense, and so forth.

They wanted to be sure that we didn't get involved in that mode because we are, after all, a regulatory agency. Furthermore, they didn't want us to build laboratories and there's a specific statement [redacted] in the conference report of [redacted] the Energy Reorganization Act of 1974, that enjoined us from doing that.

So we do our work through DOE laboratories, through contracts, through universities, companies, et cetera. We have some joint contracts with even the nuclear industry--a few, where there are special circumstances that require [redacted] that.

Now what we are trying to do, really, is not to develop a product at all but to develop ideas that can be used in setting safety requirements to affect the safety of reactors, ideas that can be used to evaluate the safety of reactors, ideas that can be used to set requirements for environmental effects and to evaluate the environmental effects, and so forth.

So our product is an idea, not a thing. [redacted] The product is not a physical thing, it's really an idea. And that really is a different kind of research than pure research which is an idea, and R&D which is a physical product. Ours is applied research in the sense that it is not pure research because it is applied to the safety of reactors and to engineering processes, but still an idea as opposed to a thing.

Now we use things in the process of generating ideas. We have to collect data; we run experiments to collect data. But the data is then compacted [redacted] into a physical model which is really an idea.

Q Does your Office of Nuclear Regulatory Research make use of data provided by currently operating reactors?

A To some extent we do, yes. Especially in our probabilistic risk assessment area, we --

Q Let's go off the record.

(Brief recess taken.)

Q I'm sorry to interrupt, Mr. Levine. Could you continue, if you can recall where you were?

A In our risk assessment area we are interested in generating failure rate data which we use in making quantitative risk assessments; for instance, assessing the reliability of sytems and the like. There has to be much more done with operating data from reactors and the Commission has recently-- I recommended to the Commission and they have recently established an operations evaluation function which will have the responsibility of doing a much more complete assessment of such data.

Q Was your recommendation a result of TMI-II of this year?

A Yes. It's been my view for several years that we have to have such a function. I tried to encourage the establishment of such a function within our Probabilistic Assessment Branch. It was fought by various people in the agency, and after TMI-II there was no fighting it any more.

Q Were your original requests in the form of documentation or letters pre-TMI?

A I guess about two or three years ago Bill McDonald and I wrote a proposal -- I forget to whom--to establish this function as a joint responsibility where we would do the technical work and his people would -- He's Haller's predecessor, Norm Haller, Head of MPA.

Q Norm Haller? MPA?

A Yes. Management and Program Analysis. And his people would do the collection and processing of the data according to models that we set up. And we were opposed in doing this, so it never bore fruit.

Q Was Norm Haller's group, if you know, responsible for the dissemination of relatively generic matters in a publication entitled, "Current Events"?

A Abnormal Occurrence Reports. I don't know what Current Events is.

Q Let me show you something that has been marked as an Exhibit to the Creswell Deposition as well as to the -- who was here last?

A I've never seen this type of document before. I don't know what it is. I don't know anything about it.

Q I would represent to you that we have been informed by other NRC personnel in the course of our depositions that this publication entitled, "Current Events Power Reactors", is produced on a periodic bi-monthly basis by the MPA group of the NRC.

A They have the basis for doing that. They publish Abnormal Occurrence Reports. They publish the compilation of LER data and there is analysis made of the LER data according to a set of guidelines to determine which are abnormal occurrences which is then reported to the Congress.

Q Do you work with those guidelines?

A I don't. My people review what they propose. I helped to generate those guidelines years ago.

Q Can we request a copy of those guidelines dealing with abnormal events be provided to us at some point?

A Haller should have them.

MR. SHIELDS: I think Norm Haller's office would have them.

MR. SIDELL: Would it be possible to get those now during the course of the Deposition? Perhaps we could make a phone call.

MR. LEVINE: I'd have to call and ask them.

MR. SIDELL: Why don't we go off the record and make a telephone call to see if we can get a copy of the Abnormal Events Guidelines.

(WHEREUPON, the Deposition recess briefly for the above-stated purpose.)

BY MR. SIDELL:

Q Mr. Levine, during an off-the-record break you have made some inquiries to obtain copies of the guidelines dealing with abnormal events and the procedures to report those events to Congress as well as a copy of a report that you produced some years ago dealing with a request to receive assistance in your research efforts in conjuncture with other offices of the NRC. Can you inform us as to what progress was made in your search for these documents?

A Mr. Hartfield is having sent down the first document. The second document you described a little incorrectly. It was a proposal that my office and another office, now known as MPA, collaborate in collecting and analyzing data, furnishing it to NRR for their use as well as our own use in research. And Hartfield said he would find that tonight and leave it with Tom Rehm tomorrow morning.

Q If we could have a stipulation that those two documents when received will be admitted as Exhibits Number 2 and Number 3, we can include them as part of the deposition.

MR. SHIELDS: So stipulated.

(WHEREUPON, the documents referred to were marked as Exhibit 2 and Exhibit 3 to the Deposition, to be furnished later.)

Q Before I continue with the substantive questions, Mr. Levine, let me briefly ask you whether or not you've been deposed before on any subject matter, not restricted to TMI?

A No. I've been a witness at several hearings, several trials.

Q Let me briefly explain then what we'll be trying to do this afternoon. Your testimony, of course, is sworn and even though we're in a conference room in one of the NRC buildings in Bethesda, Maryland, your testimony has the same force and effect as though it were given in a Court of Law before either a judge or a jury. Therefore, it is necessary that you be as precise and accurate in your responses to my questions as you can be. If you have any confusion or misunderstanding in terms of my questions, ask me for a clarification and I'll try to explain the type of information I'm looking for.

In view of the fact we have a reporter taking down your testimony it's necessary that you respond to questions audibly and without gestures of the hand or head since, obviously, those gestures are not subject to being transcribed.

Also in view of the fact that the testimony is being transcribed, it facilitates matters if you would await the completion of my questions before beginning your responses and I will likewise try and restrain myself from asking my next question until you complete your response.

At the completion of the Deposition the testimony will be transcribed and presented to you in written form for your review, correction if you feel that is necessary, and signature. You should be advised, however, that if you do make changes in the tesimony and we consider those changes to be of a substantial nature, we will be entitled to comment on those changes which may adversely affect your credibility; therefore, again, the need at this point in time to be as precise and accurate in your responses as you can be.

Do you have any questions concerning what I've just told you?

A No.

Q Getting back to the more substantive matters, in terms of your responsibilities as the Director of the Office of Nuclear Regulatory Research, have you ever had occasion to propose design changes or suggestions to manufacturers based on the research that your office has performed?

A No, I haven't.

Q Was that possibility included in your proposal of some years back?

A No. They are not related at all.

Q Have you ever had occasion to provide design changes or suggestions to utilities themselves?

A No.

Q During the time of the Three Mile Island accident in March of this year, were you involved in the hydrogen calculations?

A I was involved in trying to confirm the correctness of the calculation, yes.

Q Whose calculation were you trying to confirm the accuracy of?

A The first that I heard about the calculation it was stated that the oxygen and the hydrogen bubble in the top of the reactor vessel would be increasing at the rate of one percent per day. And I initially accepted that as being a fact that someone had calculated, but in beginning to think about it, it did not sound reasonable to me. In fact, I even had questions about whether the oxygen accumulation would be significant.

I then found out the basis for the calculation was I think a reg guide we have--I'm not sure if it's a reg guide--but we have some procedure that specifies how one should calculate the accumulation of hydrogen in the containment building given a LOCA. Of course, if you can calculate the hydrogen you can also calculate the oxygen. This ~~is~~ occurs through the radiolysis of the water by the radioactivity in the core.

That calculation is done -- excuse me-- is not applicable to the situation that occurred at TMI. It's done ~~against~~ with a nominal back pressure ~~of~~ (essentially ambient pressure in the containment) whereas the TMI reactor was at a thousand pounds pressure, and this is quite a different situation.

Q So you have a dichotomy between standard lab bases and what was actually going on at TMI in the containment, in terms of --

A Not standard lab bases. but the basis for calculating the radio-lytic de-composition rate of water, given a large LOCA in which the primary system is open to the containment is a valid calculation for that condition. It's not a valid calculation for a thousand pounds pressure in a ~~flow~~ closed system. And I was trying to track down how would one calculate that more correctly.

I spoke to a number of people and finally was quite convinced by Saturday evening that this was a gross overestimate of the situation. By Sunday noon about, after having communicated with some other people, I became convinced that there could be ~~no hydrogen accumulation in the bubble,~~ no oxygen accumulation in the hydrogen bubble in the ~~containment~~ coolant system. And there are even questions about whether that bubble was all hydrogen or not. It could have been partly steam.

Q As the basis for your conclusion that the calculations were a gross miscalculation, I believe is your terminology --

A Well, that was the final conclusion. I was not convinced in the beginning that it was a gross miscalculation. I was convinced that the conditions were grossly inapplicable.

Q And on that basis I believe one of your earlier responses was that the pressure in the containment was much greater than it would be in any other situation?

A In the primary coolant system, the pressure was much greater than it would be in the containment building.

Q And that was on the basis of a large break LOCA. Is that correct?

A The calculation that was made for the accumulation of oxygen in the containment was made on the basis of a large break LOCA. That did not exist in the TMI accident. In the TMI accident at the time this calculation was made there was no LOCA. The relief valve block valve had been closed. The system was an integral system at this point. It was at a thousand pounds pressure nominally.

Q So would the calculations then have been on a correct group of assumptions with the block valve closed?

A No, they were not. As we got more information it became clear that they were grossly inaccurate because, in fact, the oxygen, whatever oxygen could have been in the bubble would be depleted with time as opposed to increased with time in the situation that existed at TMI.

Q Whose calculations were you checking?

A I'm not sure who made the calculation. It was someone in NRR.

Q Would all of the oxygen in the containment have been a result of the LOCA in the primary system?

A Wait a minute. Are we talking about the containment or the reactor coolant system? What are we talking about? I have to understand. You have to be a little more precise.

Q Okay. I will try to be. In the containment.

A Well, the containment is normally full of air which has this, you know, 80 percent ~~oxygen~~ nitrogen and, ~~okay, or~~ SJET 20 percent oxygen, so there's a lot of oxygen in it.

(Chuckles regarding misstatement of oxygen percentage)

Sorry about that. Thank you for correcting me.

A Could I tell you --

Q Off the record.

(Brief off-the-record discussion)

Q Would all of the oxygen in the primary system have been provided prior to closing the block valve?

A You have to talk about what happened in the reactor in the two to three-hour period after the accident. When the fuel got very hot there was a metal water reaction in which the zirconium, that high temperature of the zirconium will simply take the hydrogen right out of the water and release free oxygen, and the zirconium then becomes zirconium ~~hydride~~ oxide SJET. Excuse me. It will take the oxygen right out of the water and become zirconium oxide and release free hydrogen, and the hydrogen then bubbles off into the system, and if there's a hole in the system and it gets to that hole, it will go out the hole into the containment.

Now how much got out of the containment in the two- to three-hour period, I don't know. The block valve was then closed, so there could have been some free hydrogen in the primary coolant system as a result of the metal water reaction. There should have been no oxygen free in the system. The question was: now we have a closed system where we know the decay heat is radiolyzing, or rather the decay gamma energy is radiolyzing water. So it's releasing free hydrogen and free oxygen. Is the free hydrogen and free oxygen going up into the hydrogen bubble, adding hydrogen to it and adding oxygen to it, or what's happening? That's the question. And that's the question that somebody in NRR was trying to answer. And they gave the wrong answer. The answer is that there could in no way have been oxygen added to that bubble.

Q Is there a reg guide dealing with hydrogen and oxygen production based on a small break as opposed to a large break?

A Not to my knowledge. But this is not a small break-large break question. The question is you have a closed system [redacted] and the question is are you generating free oxygen. Now in normal operation and with ~~the~~ decay heat, you're normally radiolyzing water all the time, the gamma field is radiolyzing the water, and the general way reactors operate is with a hydrogen rich water--they add hydrogen deliberately to the system so any free oxygen will be taken up by the hydrogen because they don't want free oxygen in the system because it helps corrode things, even at normal operating temperatures.

Q Do you know whether or not --

A Radiolysis is always present and it's taken care of by adding hydrogen, excess hydrogen to the water to soak up oxygen wherever it may be in the system.

Q Do you know whether there's a reg guide dealing with oxygen and hydrogen production as a result of decay radiolysis?

A I don't think so, but I'm not positive. I think not, but I'm not positive. You'd have to ask someone more qualified and with more knowledge than I have.

Q Do you know whether or not as a result of TMI-II there is a proposal to develop a reg guide for that situation?

A I don't know.

Q I believe you said previously that you were convinced on Sunday after the accident that the calculations were provided on an erroneous basis.

A Yes.

Q Do you recall who was providing you the calculations or the conclusions?

A It was a man named Schwartz from Brookhaven National Laboratories. His first name I can't think of at the moment. Did you talk to Budnitz? He may have given you his full name. Budnitz and I both talked to him.

Q And this person at Brookhaven was getting his information from the NRC or the site?

A We gave him information about plant conditions which he then used to make calculations which gave us a result that there was no possibility of oxygen addition to the bubble and if there were any oxygen in it, it would have been depleted by the radiation field.

Q When you say we gave him information --

A He had to know the size of the bubble, he had to know the temperature, he had to know the gamma flux around the fuel, and we gave him some information about that.

Q Who provided the information?

A I may have given him some and Budnitz may have given him some; I just don't remember all of it. I think Budnitz gave him the hydrogen, the gamma flux; I'm not sure.

Q Was there anyone else besides Budnitz and yourself providing information?

A I don't know. There could have been. There were all kinds of people talking to all kinds of people. I think ~~from NRR we were the principal ones;~~ from NRC we were the principal ones.

Q Do you know the time or the date that you did provide the information?

A It was either Saturday--I think it was Saturday. Could have been Sunday.

Q The day before you were convinced the information was erroneous?

A I think by Saturday afternoon late I was pretty convinced that the information was erroneous but I didn't have an alternate ~~counter~~ calculation yet, and I didn't have the results of the alternate ~~counter~~-calculation until Sunday noon, about.

Q Do you recall whether or not the conclusions provided by Brookhaven were accurate or were those erroneous calculations?

A They were accurate.

Q Who provided the erroneous calculations?

A Someone in NRR but I don't know who made the calculations.

Q Might that have been Budnitz?

A No, Budnitz -- No. He made no calculations. Budnitz is not in NRR. He's my deputy. He's in research.

Q And you can't recall who may have provided the inaccurate --

A I don't know; I have no idea who made the calculation.

Q How did you find out about the erroneous calculation?

A I called Tedesco and asked him what was the basis for the calculation and he told me. I didn't ask him who made the calculation.

Q So you found out at least what the bottom line conclusions or numbers were from Bob Tedesco?

A I got the bottom line number from someone else, but I called Tedesco to find out what the basis for the calculation was. Of cou..e, he confirmed the bottom line.

Q Of NRR?

A Yes.

Q Do you recall when that was?

A I just -- I could have been Friday night. It was more likely some time on Saturday. I would think it was on Saturday. I'm just not sure.

Q It would have to have been before Saturday in the afternoon when you became convinced that the numbers were --

A It would have had to have been Saturday morning if it was -- It couldn't have been later than Saturday morning.

Q Is there a possibility or probability that it may have been late Friday evening, for example?

A It's possible, but I just cannot recall. I just don't remember.

Q Can you recall whether or not Bob Tedesco mentioned anyone's name who may have been actually providing the data?

A I don't think he did. If he did, I can't recall.

Q At the time you found out the numbers were wrong on Friday evening-Saturday morning, sometime in that time frame --

A No, I didn't find out they were wrong then. That's when I found out the basis for the calculation and suspected they were wrong and then proceeded on my own to try to find a better way to calculate the numbers.

Q And your communications with Brookhaven then provided confirmation that the--

A I spoke first to Ritzman, Bob Ritzman, at Science Applications, Incorporated; and then later to this fellow Schwartz at Brookhaven.

Q And both of those conversations provided information to you confirming your suspicions of the erroneous numbers?

A The first one from Ritzman confirmed that my suspicions were correct but did not provide a calculational result. The second conversation with Brookhaven said yes, you're absolutely right and, in fact, he went further than I that there's no way there could be any oxygen in the bubble. And then he said if we could give him some information he would calculate what was happening. We gave him the information and he calculated it.

Q And that was late Saturday afternoon?

A It was late Saturday night or early Sunday -- No, the results of the calculation were Sunday noon-ish. The statement about there could be no way there could be oxygen in the bubble was either Saturday evening or Sunday morning and I don't remember.

Q Could Lake Barrett have been providing --

A Could what?

Q Lake Barrett, could he have been providing the calculations for NRR?

A I have no idea. I don't even know who he is. I recognize the name, but that's all I know.

Q During the time when you became convinced that the original calculations dealing with hydrogen-oxygen production were erroneous, was there any discussion of evacuation of the area?

A Would you state that again? During what time period?

Q When you became convinced that the NRR numbers were inaccurate.

A I think all talk of evacuation sort of died out by Saturday morning, and I think the evacuation scare was almost over by then.

Q Could that have been a time frame within which you would have recei[illegible] more conclusive information and therefore been the reason why the evacuation died down?

A No, no. They were not connected, in fact. I think, you know, I wasn't involved directly in that circuit so I can only give you some impressions. I know the staff recommended evacuation on Friday morning or afternoon some time. And my impression was that the major reason for that had nothing to do with the hydrogen bubble; it had to do with the fact that they didn't know the condition of the reactor and there clearly had been a large release of radioactivity from the core. And the release was so large that one would guess that the core was really not being cooled adequately at all and that one could reach the conclusion that it would be prudent to move people. And there were discussions between the Chairman and (Gov.) Thornburgh, which you can talk to them about, about what one should do.

But by Saturday morning, you know, the plant seemed to be holding and whatever suspicions there might have been about what could happen weren't happening, so the need to talk about evacuation sort of disappeared.

Q So was evacuation based on a somewhat different problem than you were concerned with?

A Well, as far as I was concerned. What do we know about what is going on in that core? We didn't. We didn't know the condition of the plant. Just ~~talk~~ think about the fact you're having a release of radioactivity from the core that's so large that you think the fuel is melting, and when fuel is melting, that threatens the integrity of the containment, you talk about well maybe we should evacuate people. But when it doesn't progress beyond that point for some hours, you know that it is not molten fuel because if it were it would have progressed. And if it is not molten fuel, you haven't got a problem that threatens the containment integrity.

Q Do you remember how large a release of radiation we're talking about now from --

A Well, I remember numbers not from the reactor to the environment; this is from the fuel into the containment building, and I remember numbers of about 30 percent of the inventory of cesium, and numbers of that order of magnitude for iodine. Those are very large releases of radioactivity. If they got into the environment, it would cause significant problems.

Q What would be considered normal releases?

A No cesium.

Q None?

A Essentially none

Q How about iodine?

A Minor traces, minute traces of iodine. When I say none, you know, insignificant amounts of cesium.

Q Were you aware of Roger Mattson's evacuation recommendation based on --

A He told me Friday afternoon some time that he had recommended evacuation. I don't know exactly when he did it, but he told me about it on Friday afternoon some time.

Q Did he indicate to you at that time that his evacuation recommendation was based on the rate of hydrogen generation?

A I can't recall very clearly, but my recollection is that it would have been based on what I just described to you as opposed to just the hydrogen level.

Q The internal releases within the containment?

A What's the condition of -- Are we really keeping that core cool? That would have been my principal concern. I think it was his, but I can't recall. There must have been about 150 phone calls that afternoon.

Q Let me go back a minute to how your research results are used. Is it fair to conclude that research results produced by your office are directed exclusively to other areas of the NRC as opposed to utilities or manufacturers?

A Yes and no.

Q Please explain.

A Our research results are published for the world at large. Every research report we issue goes into our public document room. We have extensive exchange agreements, reactor safety research exchange agreements, with over a dozen countries and we send all of our reports to these countries. So our reports are freely and openly distributed everywhere. And they are sent to all the applicable NRC offices routinely.

When we complete a significant body of research that seems to have some special moment, we write what is called a Research Information Letter and in those we state the results of the research, the basis for it, the meaning of it, and send it over to Minogue or Denton or someone, whoever it is, sometimes both.

I can give you an example. We did some research on establishing the decay heat curve for a shutdown reactor. Decay heat is a very important input to any calculation of the effectiveness of emergency core cooling systems. The less the decay heat, the less of a challenge it is to the emergency core cooling system.

The Regulatory Staff uses an American Nuclear Society curve plus 20 percent higher as the basis for their calculation of the effectiveness of emergency core cooling systems. This conservatism is inserted deliberately [redacted] in that place, as in many other places, in the model they use to evaluate the effectiveness of such systems.

One of our charters is to develop realistic models. Much more--you can never approach complete realism--but develop much more realistic models to predict the performance of emergency core coolant systems. And we did some work on the decay heat curve which showed that the ANS curve is conservative by 7 percent over the best estimate we could make.

Q How are you using the term ANS curve?

A Excuse me?

Q How are you using ANS --

A It's a decay heat curve; it's a curve for decay heat.

Q What does ANS stand for?

A American Nuclear Society. And we did some research experiments and calculations which showed that even that curve was conservative, so that the Regulatory staff is using a curve that is 27 percent more conservative than our best estimate in their calculation. And this calculation affects the prediction of peak clad temperature in a loss of coolant accident with the emergency core coolant system working correctly, and it's an important factor.

If you use just the ANS curve the prediction of peak clad temperature drops by about 300 degrees, which is a significant amount. If you use the curve we got, it would drop even more.

We wrote that up as a RIL (research information letter) and sent it to the Regulatory staff. We also sent them a RIL on the metal water reaction rate which showed that the rate they were using was conservative.

Now, I'll come back to this point, but I want to start in at it from a different direction to explain to you why we're doing such research.

There ~~is~~ was a sort of a revolt in the technical community in the early '70's about the adequacy of the research program on reactor safety under the AEC, and STET ~~the series~~ STET ~~s[illegible]~~ by people within the nuclear community, the research community, I guess, complaining about the inadequacies of the research that were being performed, and finally picked up by the Union of Concerned Scientists who wrote a report about the inadequacies of emergency core cooling system evaluations and research and so forth.

There were emergency core cooling hearings for two years starting in 1972, or maybe completing in 1972--I don't remember; around 1972.

Q Were these ~~NRC~~ AEC hearings?

A Yes. It was an ~~NRC~~ AEC rulemaking hearing, the first one we ever had, to generate a rule for the evaluation of emergency core cooling system performance. And as a result of that hearing, a rule was promulgated. There had been a rule promulgated and the rule was modified as a result of that hearing to make it better than it was.

In the meantime, the American Physical Society started a special study group on light water reactor safety and looked extensively into emergency core cooling performance, and recommended very strongly that while everyone thought there were conservatisms in the licensing model, the deliberate conservatism that I mentioned to you that were inserted into the model, they felt that it was necessary to develop a more realistic model to be sure that there was conservatism everywhere or as much as one needed.

Q Did all of the concerns you are --

A I just have to finish.

Q Okay.

A So one of the basic enterprises in our program is to develop, by means of experiments ~~in~~ and modeling, a more realistic prediction of emergency core cooling system performance. And that's why we're looking at these factors like decay heat and other things, to see how to better get data and see how to better model them.

Q Were all of the research efforts and concerns dealing with ECCS performance, assuming that the ECCS system would not run into any problems in actuation, and once being actuated whether or .ot it would perform adequately? In other words, were we all assuming the system would start and then the only question was how well would it work once it started?

A That was the conception before WASH-1400 and it was based on the use of a single failure criterion as a way of achieving enough redundancy to assure adequate reliability. But no one defined what adequate reliability was, and that's a very difficult task. I'm not saying they should have. It wasn't done and maybe doesn't even have to be done, but it wasn't done. And the reviews that were made of the adequacy of reliability in the system were done just on the basis of the single failure criterion, not on the basis of a rigorous reliability assessment. And I'm not saying that has to be done either, by the way. I'm just explaining.

Q Was there any consideration of the methods for ECCS actuation in the course of these hearings?

A Oh, yes. yes. The single failure criterion was applied to the control systems, the signals that actuated the system, as well as to the system itself, as well as to the other systems that supported the ECCS, like electric power and cooling water and so forth.

Q Was there a determination made that it would be preferable or more advantageous to have ECCS actuated on one parameter alone or divergent parameters?

A Well, you're asking me something that I have to be fuzzy on because I was in the licensing program until 1970 when I left it, and at the time that I left my concept was that one should have diverse signals. What actually happened in the bulk of the reactors, I just don't know.

Q Were there any recommendations made dealing with the ECCS hearings you previously referred to that it was better or safer to have divergent ECCS actuation rather than merely actuation based on one parameter?

A I don't know, but I would just say that the ECCS hearings were related not to reliability of operation but to the adequacy of performance given operation.

Q So there was more emphasis on how well the system would work assuming it would work in the first instance?

A How well it would perform as opposed to work, meaning operate, yes.

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Q I believe, Mr. Levine, you previously mentioned your research results were sent to several foreign countries. Is that correct?

A Yes.

Q Have you ever received information back from foreign countries dealing with specific reactor problems?

A We get research results from foreign countries. We have on occasion as an agency gotten results of events, near accidents or near misses, or what have you, at foreign reactors, yes. I'm not the principal recipient of that data. It comes in through the Office of International Programs, the information on incidents.

Q And does the information that the International Office of NRC receives then become distributed to various components within the NRC?

A Yes.

Q Would you say as a rule you receive all foreign material?

A Of interest, yes. We do not receive the equivalent of our LER's, which are component failures that are happening all the time in all things everywhere, you know--in your automobile and your telephone and everything. But we do get what you can call results of abnormal occurrences. They don't meet the same definition as our definition.

Q You anticipated my next question.

A Don't ask me what definition they do meet because I don't know.

Q Would it be closer to something where you receive information dealing with foreign events which is more on the nature of substantial LER problems, something on the order that might be produced in the Current Events publication, for instance?

A If they were to follow our definition of abnormal occurrences, we would get more than we do get. So they don't report to us lesser things. They report to us only the bigger things. I think that's what you were trying to ask.

Q Yes. Can you recall receiving any foreign information dealing with ECCS actuation?

A I don't recall any, but we may have. I just don't recall.

Q Can you recall any foreign information you may have received dealing with failures of PORV's?

A Oh, yes. I don't know if was a PORV, but a relief valve on a German reactor failed in such -- it just broke off the reactor.

Q Do you recall when that was?

A A couple years ago, about.

Q Three, four years ago?

A I think about two, but I'm not sure. We could verify the date. There is a report that we have.

Q Could we get a copy of that report?

A I can call MPA or IP.

Q Why don't we take a break and make another phone call?

A Okay.

(WHEREUPON, a short recess was taken.)

Q During our brief recess, Mr. Levine, we've made some attempts to obtain a copy of a report dealing with some foreign transients that you had received information on. Could you relate for the record what progress we've made trying to obtain a copy of that report?

A Yes, we've spoken to Mr. LaFleur and he is trying to find a copy and send it down.

Q And we have recently received copies of NUREG-0090 entitled "Report to Congress on Abnormal Occurrences, October through December 1978;"as well as copy of"Abnormal Occurrence Reports,"apparently from the Federal Register 42 FedReg 10950 February 24, 1977. Let me ask you if these are the materials you referred to that we requested earlier dealing with the abnormal occurrence guidelines?

A Yes, this document, NUREG-0090, contains Appendix A which are the abnormal occurrence criteria that we talked about before.

Q Let's mark this as Exhibit 2 to this Deposition, and we will reserve Exhibit 3 for the report dealing with the use of data from the Office of Nuclear Regulatory Research.

(EXHIBITS 2 and 3 further identified. See page 9.)

A On this other document, the Federal Register document you referred to, which also is a copy of the criteria used for selection of abnormal occurrences.

Q Is that the same criteria that is included in

NUREG-0090, or is that supplementary?

A It appears to be the same as nearly as I can tell in a quick glance. It seems to be much the same.

Q Why don't we include both of them in case there is any question subsequently, as Exhibit 2.

Have you performed any studies or research dealing with coincident logic on ECCS actuation?

A In the reactor safety study we did for two plants--the Surrey plant and the Peach Bottom-II reactor--examine and try to predict the failure probability of emergency core cooling systems to operate when needed. We had to first look at the control systems which gave them the signal to initiate operation, and in that sense we have studied whatever coincidence and redundancy there was in that logic. I can't recall.

Q Surrey is a Westinghouse facility, is it not?

A Yes.

Q Is Peach Bottom also?

A Peach Bottom is a boiling water reactor built by GE

Q And the Surrey Westinghouse facility is a PWR?

A A PWR.

Q Do you recall the approximate date of that study?

A The study started in 1972 and was published in draft form in 1974 and in final form in 1975.

Q Would it be fair to conclude that Westinghouse got a copy of that report?

A Yes.

Q Did they have any response to your research that you know of?

A I think we got comments from them, yes.

Q Do you know whether at the time they had plants set up on coincident logic for ECCS actuation?

A I don't know.

Q Do you recall the substance of the Westinghouse comments?

A No. It's all five years ago, you know.

Q What were the conclusions that your study reached?

A The study reached no conclusions. The study was an attempt to perform an assessment of the risks involved in reactor accidents in terms of the health effects on the public and damage to property. It presented the results of probability versus consequences of various sizes.

Q In your study did you deal with the divergence between pressure and pressurizer level indication in terms of ECCS actuation?

A I can't answer. I don't know. I can't recall.

Q Would that have been one of the types of problems that would have been considered?

A It's the type of problem that ought to have been considered.

Q Are you aware of a divergence between pressure and pressurizer level indication occurring at Davis-Besse on September 24, 1977?

A Generally, yes.

Q When did you first become aware of that problem?

A After TMI. Sorry about that.

Q Does your prior response that you learned about the Davis-Besse September 1977 transient post-TMI-II influence your earlier response that coincident for logic for ECCS actuation should have been considered in your earlier study?

See page 40

A No. You know, you're sort of on a track that is not logical with what we did in our study. Our study was looking at what existed, not from a licensing viewpoint, but from a viewpoint of what is the probability of things failing, and if there was coincidence, it was factored and if there wasn't, it wasn't.

Q Did your study include only domestic plants or did also include foreign?

A It included only two plants--Surrey and Peach Bottom.

Q Just Surrey and Peach Bottom. Could we get a copy both of your study as well as the Westinghouse reply comment?

A Certainly the study; I hope there is a copy of the Westinghouse comments around somewhere. I think we sent copies of the study at the request of the Commission to each of your Commissioners.

Q Well, it's undoubtedly, in the case of that situation, we have them among several other piles of paper, however.

A Well, we'll give you another --

(An off-the-record discussion held.)

Q With reference to your discussion earlier of the erroneous conclusions concerning hydrogen production, you stated you became convinced on Sunday at some point that the conclusions were --

A On Saturday night.

Q Saturday night-Sunday morning, that the conclusions were in fact erroneous. Who did you relay your conclusion to about the hydrogen production calculations?

A I think to Roger Mattson.

Q And that was on Saturday evening?

A Well, I'm not sure I told him about it Saturday evening, but I probably did.

Q Would that have been around dinner time Saturday?

A Yes, I think so, but I don't know.

Q Anyone else beside Roger Mattson?

A There may have been someone else. If he were not in the Response Center, I would have talked to whoever was there, perhaps Darryl, but I would have tried to hav[illegible] [illegible]n it to Roger Mattson.

Q That was Darryl Isenhut?

A Yes. I don't recall who I talked to. But when I did get the results of the Schwartz calculation, I told

Budnitz to call the Chairman who was then at the site at Three Mile Island and tell him, which he did. And the Chairman told him to inform the other Commissioners, which he did.

Q And this was done on Sunday?

A Sunday, around noon to two o'clock, somewhere in that time frame.

Q Were you aware of the depressurization decision occurring 7-1/2 hours into the transient at TMI-II?

A No, I didn't get told about -- I knew there was some kind of event at Three Mile Island Wednesday morning. On Thursday morning I attended a briefing of the Commission by NRR which said that everything was under control. And the next I heard about it was on Friday around noontime saying that there was deep trouble up at Three Mile Island, would I please start doing some work on it. Vic Stello called me as he was just about to leave for the site by helicopter. And that's when I knew there was deep trouble up there.

Q And at Mr. Stello's request, did you then begin to make some computations on the hydrogen production?

A No, I don't recall that he mentioned the hydrogen. I think he wanted us to work on -- Yes, excuse me. He did mention the hydrogen, but he also wanted us to work on what kind of things should we be looking at to back up the situation in case the core got into deeper trouble. This would mean, it meant to me to look at what kind of engineered safety features should we assure were operable and so forth and so on. And we began to do that kind of work, as well as look at ways to get rid of the hydrogen bubble, either by mechanical purges or by chemical absorption, and we couldn't find anything that -- We found a lot of things that would work, but they all had dangerous downsides to them and we didn't feel we could adopt any of them.

Q You first learned of the depressurization concerns on Friday?

A I don't understand the question. You'll have to be a little more explicit.

Q All right. You earlier mentioned that you knew there was deep trouble, problems at the site, and I believe that you said that was on late Thursday or early Friday?

A That was about Friday noon.

Q And at that time were you informed of the depressurization decision?

A I don't know what you mean by the depressurization decision; I'm sorry.

Q The decision evidently was made at some point in time to rapidly depressurize the primary system to try and get the system on RHR precisely.

A That was all over by Friday noon.

Q Were you involved in any way in that decision?

A No. No way at all.

Q So you merely heard about it after the fact?

A After the fact.

Q Evidently on the weekend after the accident, probably on Saturday some time, depressurization of the primary system was considered one alternative to eliminat the hydrogen in the system?

A Yes, that's correct.

Q Were you involved in those discussions?

A Yes, I was involved in those discussions to some extent. In fact, I think I was interested in exploring why couldn't we get rid of the hydrogen bubble by simply opening the pressurizer relief valve and letting it vent out. And there were concerns about where would the bubble go. For instance, as you drop the pressure in the reactor coolant system, if there was a gas bubble in the top of the vessel, it would expand as the pressure dropped; because it was 1,000 pounds pressure and if you dropped the pressure the gas expands. And then as it expanded it would perhaps go through the outlet lines, the hot leg lines, from the reactor vessel and then out the pressurizer relief valve. Or it could be swept into the steam generator. And one didn't know what would happen exactly, and one was concerned about would it expand rapidly enough to go down into the core and prevent cooling the core.

So we then thought we could do an experiment in

Idaho, in fact, on a facility we call Semi-Scale. It's a very small facility and has a reactor core about this big. But we did an experiment overnight in which we put a gas -- and it's not a nuclear core; it's an electrically heated core -- in which we put a gas bubble in the top of the vessel and we depressurized to see where the gas would go. It didn't go out the relief valve very fast nor completely. In fact, it did go over into the steam generator. So that appeared to be not a viable course of action.

Q What were the concerns if the bubble went to the steam generator?

A Well, it would interfere with circulation, as a matter of fact.

Q So it would be possible then to exacerbate--

A It could be swept into the core and affect the coolant.

Thanks, Jim.

Q We apparently have just received some more documents requested earlier.

A Yes. These are the comments from Westinghouse Corporation on the reactor safety study. It notes that we transmitted them a draft copy of the report and we asked them for comments and this is a response to our request for comments. It is dated November 1, 1974.

Q What was the problem ultimately determined to be avoided with depressurization in moving the hydrogen bubble out of the system?

A The problem -- The concern was that we were relying on main coolant pumps to remove heat from the reactor. They were operating in the environment in which they were not designed ~~not~~ to operate. They were relying on instruments for information which were not designed to operate in that environment.

Q In other words, not safety grade?

A It can be safety grade but not designed to operate in a post-LOCA environment, the safety grade equipment al. applies outside the containment, for instance. So that's a nonsequitor. But there was instrumentation that are not designed to withstand these post-LOCA environments. And the question was what's going to happen, how long will these things continue to run? Will they fail, and if they fail, what do we do? If they fail, it would be nice to get the system full of water, down in pressure and on natural circulation. But with a bubble there you are hesitant to do that. So the bubble was a kind of a burning issue to find out how to get rid of it. So we could, if we had to, go to natural circulation.

Q Was a problem dealing with the bubble the fact that it might go into the core and therefore exacerbate the uncovery problem?

A Yes.

Q Was that a primary concern, would you say?

A There was a concern that the bubble might behave in such a way as to interfere with cooling the core, whether by getting in the core or getting in the steam generator and impeding the flow, or what have you.

Q But if the bubble got into the core itself rather than the steam generator --

A The bubble would not stay in the core. I mean, gravity would make it go up, or if there was ~~flubble~~ flow it would sweep it out.

Q If the bubble got into the core as a result of the ~~re~~rapid depressurization discussion earlier 7-1/2 hours into the transient --

A Well, I can't talk -- Well, go ahead and ask your question; maybe I can answer it.

Q Would there have been a problem with further core uncovery by pushing the bubble into the reactor core as a result of rapid depressurization?

A I can only say that's a possibility. I don't know. I guess they weren't running the main coolant pumps then. If they opened the relief valve and reduced the pressure, the bubble would expand, whether it would all go out the pressurizer or not, one doesn't know; except one knows that a lot of it did because they had the hydrogen burn in the containment. So a lot of it did go out the pressurizer. Whether it further uncovered the core, now you'd have to make a very careful analysis to understand.

Q Looking back, was that one of the primary considerations that should have been avoided with rapid depressurization?

A Well, what one would have liked them to have done with the reactor was to close the block valve and turn on the HPIS and get the system solid and then they would have been fine. It would have all been over. And that could have been done any time early on before there was any damage, or even at many other times later.

Q However, we've passed this point and therefore in an attempt tc get back to a stable system --

A Now you're asking me to tell you what I think the operators were doing and I can't.

Q No, I'm asking for your opinion as to what the problems to be avoided would have been with the rapid depressuriztion decision, had it been effected.

A Well, they did try, they did open the block valve and they did try to depressurize at about eight hours into the accident. They did do that. And two hours after they opened the block valve, there was a hydrogen burn in the containment. So that was done. And we know during that time that the system was below the saturation pressure. System pressure was below the saturation pressure. So there was boiling in the system somewhere.

Q So, had they continued to depressurize the system after they in fact stopped that procedure, the results could have been much worse than the hydrogen burn?

A I guess I didn't understand that question. If they had continued -- ?

Q Their attempt to depressurize the system.

A They continued this for a long time, for several hours.

Q Would a continuation, though, after the point at which they stopped have exacerbated the problem?

A The hydrogen burn question. It's possible they could have further uncovered the core and had further metal water reaction and have generated additional hydrogen. It's possible, yes. To say whether it would have happened requires some careful analysis.

Q As part of that analysis, is core temperature an essential fact?

A That's the key. The temperature of the cladding in particular is essential.

(Deponent receives document.) Thank you.

Q I appears as though we have another document previously requested.

A This is a report of the abnormal occurrence at Gundremmingen. I guess you want me to read this and talk about it. Is that what you want to do?

Q If that would refresh your recollection.

A It would.

Q Why don't we take a break?

(A recess taken for the above-stated purpose.)

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Q During an off-the-record discussion, Mr. Levine, you stated that there was an earlier misunderstanding or some confusion about a question dealing with coincident logic for ECCS actuation, and the fact that you now believe a response you made earlier was not on the same wave length as my question. Would you care to rephrase your response? (See page 36)

A Yes, thank you. I think I misunderstood what you were saying and that's my fault, not your fault. I was really thinking of diverse signals for initiation of ECCS which could operate independently of one another in missions and not coincident signals, and I said that I had favored coincident signals and I meant to say that I favored diverse signals.

Q And diverse signals being two or more parameters each independently which would actuate the ECCS system?

A Yes.

Q And would coincident actuation be a situation where you have two parameters both of which would be required to actuate ECCS?

A Yes, that's what is normally meant by coincident and that is what I misunderstood. Now one of the diverse parameters could have been coincident. That is, one could select two signals and make them coincident as one of the set of diverse signals. That would be certainly all right.

Q Do you know whether or not currently any particular manufacturer of reactors uses diverse ECCS actuation?

A I can't answer authoritatively. I would think some do, but I just can't answer that.

Q Let me show you Westinghouse's comments which you obtained in response to an earlier request. On the eleventh page of the production which at the top has page number 4 and the title HPIS, there are three paragraphs numbered and with reference to paragraph number 3, would you review the paragraph?

Does the paragraph that you've just reviewed in the Westinghouse response to the WASH-1400 study implicitly require that there be sufficient or satisfactory indication to an operator of whether or not a LOCA exists, for example, whether the PORV's failed to open?

A I'm not sure this refers to a small LOCA resulting from a PORV being open. It could have been a pipe break. And this comment might in fact be associated with a valve in the ECCS system. Now let me read some of the other paragraphs here for a moment.

(Pause to read)

I can't be sure, but I think they are referring to valves in the emergency core cooling system and not the PORV.

Q In any event, would it be necessary in order for Westinghouse's comment there to be valid that the operator have sufficient indication to conclusively show holes in the system of one sort or another?

A You have to [redacted] have enough information to know that he was losing inventory from the system. To show holes is physically a different phenomenon that from which he would have to know he was losing inventory from the system.

Q And if we can extrapolate that to the system where we have a PORV failing open and therefore essentially a small break LOCA to the primary system without any indication available to the operator in the form of a warning light or enunciator, would it be possible then for an operator to accurately and effectively respond to that small break LOCA?

A Yes. Let's assume that -- forget a LOCA in the pressurizer at all. Let's talk about a small split in the pipe somewhere, which is a small LOCA not associated with the pressurizer. The pressurizer pressure and level signals would then be unambiguous and he would know what was going on.

Q So you are dealing with essentially indirect indicators to verify a small break?

A You can't -- There's no real way to tell you that you have a hole of a certain size in a certain location in the reactor coolant system. You can tell from plant parameters, from monitors which measure radioactivity in the containment, and so forth.

Q Well, currently on most --

A The problem of the water discharge, if it's through the relief valve from the water going ~~through and~~ through the tail pipe.

Q Well, dealing with a failed open PORV on most B&W reactors, there apparently is at least an indirect warning light showing that a signal has been sent to the solenoid to energize the PORV to close, not necessarily an actual position indication.

A That's correct.

Q That is one relatively direct method available to an operator to determine if the PORV is open or closed. Correct?

A It's in fact indirect.

Q It's indirect but it deals directly with the PORV as opposed to a plant parameter?

A It deals directly with the PORV but it is classically known to be a bad way of measuring valve position.

Q When was the first time it was known to be a classically bad example of measurement? Is this pre-TMI?

A I've known for years. It's the kind of thing [redacted] where you try to measure something by measuring another parameter when you can measure the parameter of interest directly. And I've heard of examples for years of people measuring valve positions by measuring the actuator signals as opposed to the actual position of the valve.

Q Do you know whether or not it was common knowledge or a common position in the NRC that it was a classically bad situation to have an indirect PORV indicator?

A I don't know what's common in NRC. This has been my experience.

Q Did anyone beside yourself in the NRC have the same opinion as to the quality of the indirect measurement of the PORV that you know of?

A I have to correct your question a little bit. I'm not talking about the PORV. If you talk generally, forgetting the PORV, I think you will find that a number of people will tell you that's not a good way to do things; one should do them by measuring the actual valve stem position.

Q The same kind of problem for a B&W pressurizer providing indirect measurement of core inventory?

A I guess I'm not sure that I understand your question.

Q Well, you indicated earlier that at least in dealing with a PORV or generally any kind of mechanism, it was always better to have --

A Not in dealing with a PORV specifically but generally with valves, yes.

Q -- that it would be better to have a direct rather than an indirect measurement?

A Yes.

Q Or alternatively it would be better to have a measurement directly from the particular item or mechanism you wanted measured rather than having one mechanism measure what was going on in the second mechamism?

A Yes.

Q In a B&W reactor the pressurizer level indication measures indirectly what's going on with core inventory, does it not?

A It gives you some indication of inventory, yes.

Q But it's not the same as though you would have thermocouples, for instance, in the reactor core itself providing direct information as to what's going on in the core?

A Or water level indicator or an indicator that measured boiling or something, that's correct.

Q Pre-TMI-II, do you know whether or not it was technologically feasible to have a direct water indication in a B&W reactor?

A Why it's technologically feasible to measure water level in the pressurized water reactor; you'd have to decide exactly what you want to measure and for what purpose. You may have to develop the instrument, but I'm sure you could.

Q Are we talking about a relatively substantial cost involved in something like that?

A It depends on what you want. If you want to measure whether there is a bubble in the top of the reactor vessel, that could be done quite easily. If you also want to measure ~~is~~ a bubble on top of the steam generator, you could do that. That's another instrument. If you want to measure bubbles in the core, that's a much more complex thing, but it could probably be done also. Do you want to measure boiling in the core, which are bubbles, which are gas bubbles, water or steam?

So my recommendation to this agency will be--in fact, it's in my budget--to make studies of what instruments, not just this area but for the whole reactor, what instruments are needed to follow the course of accidents, a very thorough study in terms of what do you want to measure, what do you need to know to define what's going on, how you're going to measure it, what are you going to do with the information, and so forth? And we plan to make such studies if we get the proper funding, and I think we will.

Q Do you know what the reason for allowing an indirect indication on a PORV w s pre-TMI-II?

A No, I don't know.

Q Could it have been the fact that the PORV itself was not defined as safety related?

A That's possible, but I don't know. I'm really the wrong person to ask that question.

Q Let's mark the Westinghouse reply comments to WASH-1400 as Exhibit 4 to the Deposition, reserving as Exhibit 3 the two-or three-year old report or request for your data assimilation. Do we have a name that we can refer to that, as the Levine study memo?

A The Levine/McDonald memo.

Q Okay. The Levine/McDonald memo will be admitted as Exhibit 3 when we get a copy; and let's have this as Exhibit 4, which is a November 1, 1974 letter from R. Salvatori, Manager, Nuclear Safety Department of Westinghouse Corporation, Power Systems, addressed to Mr. Saul Levine at that time at the U.S. Atomic Energy Commission, and attached comments.

(WHEREUPON, the document referred to was marked as Exhibit 4 to the Deposition.)

Q Earlier, Mr. Levine, I asked you whether or not you had any information from foreign transients dealing with PORV problems, failing open or general PORV problems, and you indicated that in fact you did have information dealing with a specific reactor in Germany.

A Yes.

Q Have you received any additional information that would assist you in recalling the specifics of that incident?

A Yes, I have.

Q First of all, let me ask you what documents you have received?

A I have received three documents. One, a memo to file, Nuclear Regulatory Commission, dated January 19, 1977 written by Joe LaFleur. The other is an unidentified document just titled "Abnormal Occurrence at the Nuclear Power Plant Gundremmingen, January 13, 1977" which has attached to it a report to the Bavarian Parliament, a draft dated January 19, 1977, informal translation. And a third document which is a Department of State telegram from the American Embassy Bonn, Germany, to the Secretary of State addressing the incident at Gundremmingen Nuclear Power Station, which happened on January 13, 1977.

Q Why don't we mark those as Exhibits 5 through 7, respectively to this Deposition and then we'll give them back to you so that you can refer to them.

(WHEREUPON, the documents referred to were marked as Exhibits 5, 6, and 7 to the Deposition.)

Q Having had the opportunity to review Exhibits 5 through 7 dealing with the German transient and the PORV problem, what can you tell us now about that particular incident?

A Yes, I reviewed the three documents I mentioned before and what happened was some kind of a transient in the electrical system that the reactor plant was supplying and the failure in the control of the turbine which made it appear that additional steam flow from the reactor was required and that steam flow was supplied (which in a boiling water reactor drops the level of water in the reactor markedly) and the plant has to respond by supplying more feedwater, which it did. And the feedwater valve stuck open so that it kept supplying water even after the transient which needed the additional steam was over. It kept supplying a large amount of feedwater and it resulted in overpressurizing the reactor and the 14 relief and safety valves all opened and one relief valve, because of the physical reaction forces of the steam & water flow through the valve, bent. The pipe that attached to the reactor vessel bent and split open, but the valve remained attached to the reactor.

Q Was this essentially a case where the steam pressure was just greater than had been considered in the design of the PORV and bent it?

A Well, the safety and relief valves are provided just for this purpose--to prevent overpressurizing the vessel. So they are there for that purpose.

Q But apparently the --

A It's an abnormal condition, but the American Society of Mechanical Engineers code in our country requires relief and safety valves and it is a practice followed all over the world.

Q But apparently the pressure involved in this particular reactor was of such a high level it did more than merely open the valves--it bent one of them.

A No, the pressure didn't bend the valve. It was when you shoot a steam jet or steam and water out of a valve, there's a reaction force. It is what makes rockets go. And the valve wasn't designed to take that reaction force, so the pipe bent and split open part way around.

Q Is there any indication whether there was an indicator light or some kind of information that was provided to the operator as to the particular problem?

A I saw nothing in here that discussed that matter.

Q Can you recall any other foreign problems dealing with failed open PORV's?

A I don't know of any. That doesn't mean there have not been any. But I don't know.

Q Do you know who in the NRC decides what is safety related?

A It's decided in NRR principally, and to some extent in Standards, Office of Standards Development.

Q Do you know what the basic definition of safety related for reactors is?

A It starts with a general design criteria which talks about achieving safe shutdown of the reactor and coping with accidents. That is if we postulate a series of design basis accidents and those systems that are installed in the plant to cope with those design basis accidents are safety related.

Q So if something was not a design basis accident, in other words, beyond the bounds of the analysis originally, it would not be considered safety related?

A That's a generalization which is true, but there could be some exceptions.

Q Well, as a consequence of a failed open PORV, you have a small break LOCA in the primary system.

A Yes.

Q And is it your opinion that that situation can affect the safe shutdown of the plant, provided you have either no indication or an indirect indication that the PORV has failed open?

A It doesn't matter what the indication is ~~that can~~ [redacted] a stuck open valve that puts you in a small LOCA and you have to handle the situation.

Q But the operator can much more accurately and easily handle the situation if he's got an indication?

A Yes, that's exactly right.

Q That also happens to be accurate.

A Yes, of course.

Q Pre-TMI-II this year, were you aware of any other domestic PORV problems at B&W facilities?

A No.

Q Are you aware of any now?

A Yes.

Q Which ones?

A Davis-Besse and Rancho-Seco.

Q In particular, Davis-Besse, any date or just a problem at Davis-Besse?

A It happened much earlier than TMI. I don't remember the exact date.

Q Were you aware of any cases, pre-TMI again, of pressurizer level indication going off scale high?

A No.

Q Would that be construed as a relatively exceptional or unusual occurrence?

A I don't know.

Q Is the pressurizer level indication a safety related piece of a B&W reactor?

A As I understand it, the licensing people did not consider it to be so, but I'm not sure of that.

Q When we first began the Deposition you indicated your office received selected LER's dealing with relative information for your office.

A No, no. We get all the LER's.

Q Okay. Did you, if you know, receive the LER's on, for instance, Davis-Besse in September 1977?

A I would think so, but I don't know as a fact.

Q Do you have any organizational structure that in reviewing LER's makes a determination that there may be a potential generic safety problem?

A No. I told you before, we review them for failure rate date on components and systems, but not for safety significance.

Q Well, if you have a valve, for instance, that has a high failure rate and might not be classified by the strict definition as safety related but can produce safety related consequences --

A You're thinking that I'm in the licensing business and I'm not. I'm in the business of doing research and one of my jobs is to get prepared to better risk assessments than we did on WASH-1400. And obviously one needs as big a data base as one can get. The principal focus of our efforts in regard to LER's and also ~~M~~NPRDS, which is an industry system to take that data, and analyze it for failure rate data—component failure rate data. We are looking at differences in data between data we used in WASH-1400 ~~on~~ and what exists now. We are looking at the differences in data among plants, which there are. But it is not our charter to determine the safety significance of those things.

Safety significance reviews are done in the licensing area.

Q As part of your Office's responsibilities for making proposals for design changes --

A I don't have any responsibility for proposing design changes. I have the responsibility to do research, for instance, on improving the safety of reactors. When that research is done, it will be utilized by NRR and SD to determine if they want to make additional safety requirements on the industry.

Q Well, in the course of that particular research is it not within that realm of responsibility to determine the risks involved of safe plant shutdown from an accident which might be induced possibly by a failed open PORV which goes unnoticed?

A No. By the same token I would say if anything came to our attention and it was clearly a matter of safety significance, that we recognize it as a matter of safety significance, we are bound to inform NRR and we do.

'ape 5

Q Did you perform any studies dealing with PORV's pre-TMI-II?

A We had a failure rate, we had a probability of sticking open for PORV's, yes, which we used in WASH-1400.

Q Was that exclusively part of WASH-1400?

A Yes.

Q Where did that study go?

A It came from reviewing many sources of data, not just nuclear power plants, PORV's in many plants, many kinds of plants. And we determined from that a failure, a stick open probability for such valves, which was about 1 in 100. As it opened, it had about 1 in 100 chance of failing to reclose.

The latest data from operating reactors is 1 in 50, so we bring in a factor of 2, which is very good for this business, by the way.

Q Was the PORV study you just referred to broken down by particular types of plants; for instance, a B&W PWR?

A No.

Q You just mentioned that the failure rate was 2 out of 100 for PORV's failing to reclose.

A For B&W PORV's.

Q Two out of 100?

A Two openings out of 100 would fail to reclose.

Q Are you aware at this point in time that there have been five PORV's failing to close on B&W reactors?

A I think that's where this number comes from, yes.

Q So you're saying that of the five B&W PORV's that have failed to close, that was based on 250 openings?

A Excuse me. I guess I recall we had three failed out of 150 openings, and that's what my recollection is. Now if there are five, that's another factor a little bit.

Q Well, would 5 out of 150 --

A That would be 1 out of 30 instead of 1 out of 50.

Q Would that significantly change the --

A It doesn't change things very much in the parameter of this. You're really looking for factors of 5 and 10 to ~~be~~ define significant difference.

Q So 1 out of 30, 3 percent failure rate, would not be significant in your opinion?

A No, no, no. I didn't say that. I said the difference between 1 in 50 and 1 in 100 and 1 in 30 is not very significant. The difference between 1 in 10 and 1 in 100 is quite significant in probability terms.

Q Were any studies dealing with loss of pressurizer level indication performed by your office?

A Not that I'm aware of.

Q Let's go off the record for a minute.

(A brief recess was held.)

Q Mr. Levine, what kinds of research efforts are produced by your office?

A Research results are produced to encompass the entire nuclear fuel cycle from reactor safety issues, to environmental issues, to improve risk assessment techniques. These results are used in various ways. They can be used just to add to a store of information. They can be used to modify rules and regulations, safety guides. They can be used to solve generic issues. They can be used in a variety of ways. They close doors sometimes. There's an open question; it just simply gives you an answer and says don't worry about that any more, which doesn't change anything in the licensing process but it is useful to know you don't have to worry about that any more.

Q Are your research results made available for use in licensing?

A Yes.

Q Routinely?

A Routinely.

Q As well as to operating reactors?

A Our reports are published; they are available to anyone.

Q So an operating reactor could consider the information for whatever method they chose?

A Yes. But mostly our results are not directly applicable to operating problems; they're more directly applicable to analysis problems which go more to reactor vendors and architect-engineers as opposed to operating utilities. But in some cases we have produced results that will directly affect ~~the design,~~ design changes to utility plants.

Q Can you give me a specific example as to a design change?

A Yes. We have just reviewed the 24 different varieties of auxiliary feedwater systems in operating PWR plants built by Combustion Engineering and Westinghouse. And we have evaluated their relative reliabilities and we find there is a wide variation in reliability and that some of those systems need fixes. We have made recommendations to NRR as to what fixes are needed to improve the safety of these systems. This is the first time we've done this kind of an exercise, by the way. And they are in the process of implementing those recommendations.

Q Do your research results involve any changes in training procedures?

A They haven't yet but they could ultimately. Let me say first of all that WASH-1400 found that there were -- we have to operate with human error and we had to assess the possibility of human error, and in doing so we looked at operating procedures and maintenance procedures, test procedures and emergency procedures. And we found some difficulties with those procedures.

But we found in talking to the plant people that they were aware of those deficiences and they were doing things in a more rational way. We based our assessment on the way they were actually doing them.

For the future we feel that we need to study the matter of operator training in the following way: We need to know what data the operator needs to follow the course of accidents, and I mentioned before, we are going to do studies in that area. We know the way in which that information should be displayed for maximum effectiveness to aid the operator. We think that there should be diagnostic equipment to help him decide what all this information means. We think we have to understand better the kind of transients and events and small LOCA's that can lead to things like TMI, and we're going to develop models to try and predict that better, ~~and~~ These models can then be the bases for making requirements for training simulators that go beyond design basis accidents. We think that simulators should be changed as soon as we can state the requirements; then we can go beyond design basis accidents in training operators.

Q In your response you indicated you'd spoken with reactor operators to find out some of their specific procedures.

A Yes. This was during WASH-1400. That was about five or six years ago.

Q At that time was there any reliance by operators on pressurizer level indication as a primary indicator of core inventory discovered?

A I don't know.

Q If it was, it would be included in WASH-1400?

A If it was, it would be included. I don't know whether we considered it or not.

Q Does your office have any specific role in terms of plant licensing aside from the provision of research results to licensing, as previously mentioned?

A No.

Q Do you have any involvement in terms of your research results with operating reactors?

A Only in the sense that when we produce a research result that's of significance we discuss it with NRR and talk to them about methods to implement and whether it should or not be implemented. But it is their action.

Q Does your office engage in any data reviews or analyses or verification of particular events relative to the use of simulators?

A No.

Q Do you have any involvement --

A We're planning to make studies of how to improve simulators.

Q Currently your office has no dealings relative to simulator use?

A No.

Q What particular relationships between your research results and SD exist?

A Much like with Licensing. For instance, this result I talked to you about on decay heat would ultimately end up in a change to our regulations which would be written by Standards, but the writing, the decisions, and the discussions would involve NRR and ourselves and Standards.

Q What about your office's relationships w.th NMSS?

A We do research that will produce results that they can use, too, on nonreactor facilities--waste storage facilities and the like.

Q What about I&E?

A We have a lesser interaction with them, although we are doing some work for them to try to indicate the relative importance of their inspection modules and also do some work on how they can best utilize their on-site inspectors. These are programs we've just started.

Q Pre-TMI did you have any input from I&E in terms of whether or not they might spot a particular generic problem and want you to look into it?

A No.

Q What kind of involvement does your office have with ACRS, if any?

A Significant. They are required by the Congress to review our research program annually and have done so for the past two years. This year they are required by the Commission to review our budget and advise the Commission on our budget, which they have just done. This means there are a series of meetings that last throughout the year reviewing the details of our program and our future programs.

Also they ask us for advice on issues from time to time.

Q Pre-TMI-II do you know whether or not your office received a copy of the Michelson Report?

A I think not.

Q Have you seen a copy of that since?

A I have seen a copy since.

Q Have you read it?

A I read it briefly, yes.

Q Do the concerns raised in the Michelson Report appear to be the type of matters your office would get involved with?

A Not in the past.

Q Evidently in the present and future?

A Well, it's not clear. You know we've just established this new Operations Evaluation function which will have the charter of reviewing the safety significance of failures, but there's a little satellite group in each office that will interact with that centralized group and just how the job will be split up, I don't know. We may be involved in reviewing the safety significance. I just don't know.

Q What relationship does your office have with EPRI?

A We meet with them about quarterly and review their programs and our programs, and suggest programs that they should do and they suggest programs that we should do, and we conduct some joint programs where there is funding by industry and by EPRI and by ourselves, in a few cases. There are not many of those but there are a few.

Q Beside providing copies of your research results to utilities, do you have any involvement with utilities directly?

A Almost none.

Q What about with the Department of Energy or other Federal agencies?

A Yes. We have a significant involvement with other agencies. We have other agencies doing research for us that we manage. Of our budget, 85 percent is spent in DOE laboratories. For ~~A~~ lot of it, we have also contracts with the Geological Survey, the Coast and Geodetic Survey, with HEW and EPA and others. So we have an extensive interaction with other Government agencies in terms of getting our research done.

Q Do they work on particular types of projects?

A Yes.

Q Which particular kinds?

A Well, almost all our reactor research is done at DOE laboratories. Our safeguards research is done there. The other agencies work on problems in which they have specialties such as seismic, tornado, floods, et cetera, health effects of radioactivity.

Q How has the NRC used what is known as either WASH-1400 or the Rasmussen Report?

A It's hard for me to summarize coherently how it has been used. You will find that NRR view is that while they have occasionally used its techniques, they have not relied on these techniques very heavily. It's my opinion and has been for some years pre-TMI, that there should have been much more extensive use of these techniques to help solve licensing problems where you can't make a decision clearly without this. It will certainly almost always lend some insight in most problems, but not all; not all problems can benefit from this technique.

I think the Commission's policy statement on the Lewis Report dampened the application of these techniques. But on the other hand, I think the Commission wants these techniques used more and more. I have a very clear signal on that, and it's happening; it's beginning to happen. This auxiliary feedwater study I mentioned, for instance, is a very good example of that.

To enhance the applicability of these techniques, one has to train people on how to use them. They are quite subtle. It's very easy to make mistakes with them and we have to develop a cadre of skilled practitioners in the agency and in laboratories and elsewhere. It's a new methodology.

Q Who in the NRC is the central point of information for WASH-1400 concerns, if there is one?

A My -- the Probabilistic Analysis staff.

Q Is that exclusively within your office?

A You'll find everybody willing to express opinions on it.

Q Post-TMI.

A Pre-TMI and post-TMI both. I think TMI in fact enhanced the importance of WASH-1400 in the eyes of many people because they found that accident sequence very similar in WASH-1400 as to what happened at TMI; not the same sequence because B&W sequence is different from the Westinghouse sequence. But it was there. It pointed out the fact that a stuck open relief valve could occur and it could lead to a small LOCA.

Q Is it your opinion that the Rasmussen Report or WASH-1400 should be reevaluated in view of TMI-II?

A No, it should not. I think, again, you have to understand that WASH-1400 was a risk assessment and not a licensing tool. The methods within it can be used to aid licensing but are not a replacement for many of the kinds of analyses that are now performed. To update WASH-1400 in a significant way requires the development of significant methodological improvements in a few areas, and that's going to take a while to do and then it could be updated.

But the accident sequence that occurred is there. It is essentially there.

Q What particular areas?

A They are enumerated in the Lewis Report and they are the areas that we told Lewis that needed improvement. We need a better seismic risk model, tornados, floods, and so forth, many things like that; we need a better data base.

Q Any materials or areas internal to the reactor itself as opposed to the external ones you just mentioned?

A Better data base, better models for some human operator openings, and so forth. Those are internal. But mostly external.

Q Can we get a copy of the Lewis Report?

A Sure.

Q Is one available immediately, or if it is not convenient we can just have it sent to us.

MR. SHIELDS: I'm sure you must have been sent copies of that along with WASH-1400.

MR. LEVINE: ~~Gordon Strap~~ B.ll Stratton has probably got it.

MR. SIDELL: Okay.

MR. LEVINE: If you haven't, I'll send you as many as you want.

MR. SIDELL: Mr. Shields, do you have any questions?

MR. SHIELDS: I have no questions.

MR. SIDELL: At this time, I've run out of questions of mine and others. In view of our prior policy in terms of what we're doing with Depositions, we will recess rather than adjourn the Deposition, should in the unlikely, hopefully, event we have more, we can more easily contact you and continue on. I will tell you that we have not recalled anyone for a Deposition so far, although we plan to do it in a rather small finite number of cases. I would doubt that that would be the case with you, although I can't make any promises in that regard.

But at this time we will consider the Deposition in recess. We certainly thank you for your patience and your assistance.

MR. LEVINE: Thank you. I'm happy to help.

(WHEREUPON, at 5:35 p.m. the Deposition was recessed.)

- - -

I have read the foregoing pages, 1 through 73, and they are a true and accurate record of my testimony therein recorded.

Saul Levine

SAUL LEVINE

Subscribed and sworn to before me

this____________day of________,1979.

____

Notary Public

My Commission Expires:______________

REPORTER'S CERTIFICATE

DOCKET NUMBER:

CASE TITLE: DEPOSITION OF SAUL LEVINE

HEARING DATE: August 8, 1979

LOCATION: Bethesda, Maryland

I hereby certify that the proceedings and evidence herein are contained fully and accurately in the notes taken by me at the hearing in the above case before the PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND and that this is a true and correct transcript of the same.

Date: August 9, 1979

Official Reporter
Acme Reporting Company, Inc.
1411 K Street, N.W.
Washington, D.C. 20005

CERTIFICATE

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

Date: August 14, 1979

Saul Levine

Saul Levine

Exh #1

BIOGRAPHICAL DATA

SAUL LEVINE

Education:

1945 - B.S., U. S. Naval Academy
1949 - B.S. in Electronics, Massachusetts Institute of Technology
1955 - M.S. in Nuclear Engineering, Massachusetts Institute of Technology

Positions Held:

1945 - 1954	U. S. Submarine Service
1955 - 1958	Project Officer U.S.S. Enterprise, under ADM. Rickover. Responsible for directing all technical, financial, production and administrative aspects of the reactor plant prototypes and the production plants for the world's first nuclear powered aircraft carrier, (U.S.S. Enterprise).
1958 - 1962	Polaris Missile System under ADM. Rayborn. Managed design, integration, installation testing and performance evaluation of the Polaris Navigation System.
1962 - 1970	Assistant Director for Reactor Technology in the Division of Reactor Licensing, USAEC. Responsible for directing the development of nuclear safety review techniques for nuclear reactors, requirements for safety research and development, and the technical safety reviews of reactors of all types.
1970 - 1972	Assistant Director, Division of Environmental Affairs, USAEC. Managed programs related to environmental impact associated with AEC's programs and assisted in the establishment of requirements for the implementation of NEPA in the AEC.
1972 - 1975	Project Staff Director of the Reactor Safety Study, USAEC. With Professor Rasmussen of MIT, provided the principal technical and management direction of the study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants."
6/73 - 1/75	Special Assistant to the Director, Division of Reactor Safety Research, USAEC.
1/75 - 1/77	Deputy Director, Office of Nuclear Regulatory Research, USAEC.
1/77 -	Director, Office of Nuclear Regulatory Research, USNRC. Management of research program to confirm assessments used by the Commission in regulating the commercial uses of nuclear energy - particularly in the areas of nuclear safety, safeguards and environmental impacts.

Levine Deposition Exh. #1

Federal Register (42 FR 10950) February

ABNORMAL OCCURRENCE REPORTS

Implementation of Section 208, Energy Reorganization Act of 1974; Policy Statement

BACKGROUND

Section 208 of the Energy Reorganization Act of 1974 (Pub. L. 93-438, 42 U.S.C. 5848) provides that:

> The Commission shall submit to the Congress each quarter a report listing for that period any abnormal occurrences at or associated with any facility which is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954 as amended, or pursuant to this Act. For the purposes of this section, an abnormal occurrence is an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety. Nothing in the preceding sentence shall limit the authority of a court to review the determination of the Commission. Each such report shall contain—
>
> (1) The date and place of each occurrence;
>
> (2) The nature and probable consequence of each occurrence;
>
> (3) The cause or causes of each; and
>
> (4) Any action taken to prevent reoccurrence.
>
> The Commission shall also provide as wide dissemination to the public of the information specified in clauses (1) and (2) of this section as reasonably possible within fifteen days of its receiving information of each abnormal occurrence and shall provide as wide dissemination to the public as reasonably possible of the information specified in clauses (3) and (4) as soon as such information becomes available to it.

On March 17, 1975, the Commission published a notice in the FEDERAL REGISTER "Reporting of and Dissemination of Information Concerning Abnormal Occurrences." (40 FR 12166). The notice stated that the Commission has under active consideration the formulation of proposed amendments to its regulations to facilitate implementation of Section 208 of the Energy Reorganization Act of 1974 and that an appropriate notice of proposed rulemaking would be published in the FEDERAL REGISTER for public comment before any amendments are adopted. Since implementation involves the conduct of Commission business and does not impose requirements on licensees, the Commission is not proposing amendments to its regulations but is instead issuing a general statement of policy.

In July 1975, in the exercise of the authority conferred upon it by Congress to determine which unscheduled incidents or events are significant from the standpoint of public health or safety and are reportable as abnormal occurrences, the Commission developed interim criteria[1] for evaluating licensee events. On the basis of these criteria and as required by Section 208, the Commission has issued five quarterly reports to Congress on abnormal occurrences. The five reports are: (1) NUREG 75-090, Report to Congress on Abnormal Occurrences, January-June 1975, dated October 1975; (2) NUREG 0090-1, Report to Congress on Abnormal Occurrences, July-September 1975, dated March 1976; (3) NUREG 0090-2, Report to Congress on Abnormal Occurrences, October-December 1975, dated March 1976; (4) NUREG 0090-3, Report to Congress on Abnormal Occurrences, January-March 1976, dated July 1976; and (5) NUREG 0090-4, Report to Congress on Abnormal Occurrences, April-June 1976, dated October 1976. These reports are available from the National Technical Information Service, Springfield, Virginia 22161.

Based on its experience to date in the preparation and issuance of abnormal occurrence reports, the Commission has decided that its responsibilities under Section 208 can be carried out more effectively if the interim criteria now used to identify abnormal ocurrences are further refined. Accordingly, the Commission is issuing this general statement of policy which describes the manner in which the Commission will, as part of the routine conduct of its business, carry out its responsibilities under Section 208 of the Energy Reorganization Act of 1974 for identifying abnormal occurrences and making the requisite information concerning each such ocurrence available to the Congress and the public in a timely manner. Included in the policy statement are revised criteria which the Commission will use in determining whether a particular event is a reportable abnormal occurrence within the meaning of Section 208. It is expected that as additional experience is gained, further changes in the criteria may be required.

ABNORMAL OCCURRENCE CRITERIA

The criteria contained in the general statement of policy have been developed to comply with the legislative intent of Section 208—to keep Congress and the public informed of unscheduled incidents or events which the Commission considers significant from the standpoint of public health or safety. The criteria reflect a range of health and safety concerns and are applicable to events involving a single occupational worker as well as those having an overall impact on the general public.

The criteria establish a threshold for reporting. Occurrences that meet or exceed the threshold wil be reported as abnormal occurrences. The Commission has established the reporting threshold at a level which will assure that all events likely to be of significance from the standpoint of public health or safety will be reported. At the same time, by placing the reporting threshold generally above the level of events required to be reported to the NRC, the Commission will not report for Section 208 purposes those events reported by licensees which involve some variance from regulatory limits but which are not significant from the standpoint of public health or safety.

LICENSEE REPORTS

This general statement of policy will not change the reporting requirements imposed on NRC licensees by Commission regulations, license conditions or technical specifications. NRC licensees will continue to submit required reports on a wide spectrum of events, including such events as minor instrument malfunctions and slight deviations from normal operating procedures which are without significance from the standpoint of the public health or safety but which provide data useful to the Commission in monitoring operating trends of nuclear power facilities and in comparing the actual performance of these facilities with the potential performance for which the facilities were designed. In accord with present policy, information relating to all events reported to the NRC will continue to be made available to Congress and placed in the NRC Public Document Rooms for public perusal. Information can also be obtained by writing the Nuclear Regulatory Commission, Public Document Room, Washington, D.C. 20555. In addition, the Commission will continue to issue news announcements on events that seem to be newsworthy regardless of whether or not the events are designated abnormal occurrences.

The Commission invites all interested persons who desire to submit written comments or suggestions on the abnormal occurrence criteria in this general statement of policy and on the examples of abnormal occurrences in Appendix A thereto, to send them to the Secretary of the Commission, United States Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch by May 25, 1977. Consideration will be given to such submissions in connection with possible future revision of the criteria. Copies of comments received by the Commission may be examined at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C.

GENERAL STATEMENT OF POLICY ON IMPLEMENTATION OF SECTION 208 OF THE ENERGY REORGANIZATION ACT OF 1974, AS AMENDED

1. *Applicability.* Implementation of Section 208, Abnormal Occurrences Reports, involves the conduct of Commission business and does not impose requirements on licensees. Reports will cover certain unscheduled incidents or events related to the manufacture, construction, or operation of a facility or conduct of an activity subject to the requirements of Parts 30, 40, 50, 70 or 71 of Chapter I, Title 10, Code of Federal Regulations.

2. *Definition of terms.* As used in this statement: (a) An abnormal occurrence subject to the provisions of Section 208 of the Energy Reorganization Act of 1974 means an unscheduled incident or event at or associated with any activity or facility which is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954, as amended, or pursuant to the Energy Reorganization Act of 1974, which the Commission determines is significant from the standpoint of public health or safety.

3. *Abnormal Occurrence criteria.* The Commission will apply the following criteria in determining whether an event at a facility or involving an activity licensed or otherwise regulated by the Commission is an abnormal occurrence within the purview of Section 208 of the Energy Reorganization Act of 1974. Events determined to be at or above the threshold

[1] The interim criteria are set out in an appendix to each report to Congress on abnormal occurrences.

POOR ORIGINAL

established by the criteria will be subject to the reporting and public information requirements of Section 208 of the Energy Reorganization Act of 1974.

(a) An event will be considered an abnormal occurrence if it involves a major reduction in the degree of protection of the public health or safety. Such an event would involve a moderate or more severe impact on the public health or safety and could include but need not be limited to:

(1) Moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the Commission;

(2) Major degradation of essential safety-related equipment; or

(3) Major deficiencies in design, construction, use of, or management controls for licensed facilities or material.

Examples of types of events which might be determined to be abnormal occurrences in accordance with these criteria are set out in Appendix A of this general statement of policy.

4. *Commission dissemination of abnormal occurrence information.* (a) The Commission will provide as wide a dissemination of information to the public as reasonably possible.[2] A FEDERAL REGISTER Notice will be issued on each abnormal occurrence with copies distributed to the NRC Public Document Room and all local public document rooms. When additional information is anticipated the notice will indicate that the information can be obtained at the NRC Public Document Room and in all local public document rooms.

(b) Each quarter, the Commission will submit a report to Congress listing for that period any abnormal occurrences at or associated with any facility or activity which is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954, as amended, or pursuant to the Energy Reorganization Act of 1974, as amended. This report will contain the data, place, nature and probable consequence of each abnormal occurrence the cause or causes of each abnormal occurrence and an action taken to prevent recurrence.

APPENDIX A—EXAMPLES OF ABNORMAL OCCURRENCES

Examples[3] of types of events which might qualify as abnormal occurrences under the criteria in paragraph 3 of the General Statement of Policy are listed. These examples are hypothetical only. Whether a particular event will be determined to be an abnormal occurrence will depend on the specific facts and circumstances of the event.

I. *For all Licensees:*

A. Human Exposure to Radiation from Licensed Material:

1. Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual to 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation (10 CFR 20.403(a)(1)), or equivalent exposures from internal sources.

2. An exposure to an individual in an unrestricted area such that the whole body dose received exceeds 0.5 rems in one calendar year (10 CFR 20.105(a)).

B. Discharge or Dispersal of Radioactive Material from its Intended Place of Confinement:

1. The release of radioactive material to an unrestricted area in concentrations which, if averaged over a period of 24 hours, exceeds 500 times the regulatory limit of Appendix B, Table II, 10 CFR Part 20 (10 CFR 20.403 (b)).

2. Radiation or contamination levels in excess of design values on packages or loss of confinement of radioactive material such as: (a) a radiation dose rate of 1,000 mrem per hour three feet from the surface of a package containing the radioactive material, or (b) release of radioactive material from a package in amounts greater than the regulatory limit (10 CFR 71.36(a)).

C. Theft, Diversion, or Loss of Licensed Material, or Sabotage or Security Breach:[4]

1. Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas.

2. A substantiated case of actual or attempted theft or diversion of licensed material or sabotage of a facility.

3. Any substantiated loss of special nuclear material or any substantiated inventory discrepancy which is judged to be significant relative to normally expected performance and which is judged to be caused by theft or diversion or by substantial breakdown of the accountability system.

4. Any substantial breakdown of physical security or material control (i.e., access control, containment, or accountability systems) that significantly weakened the protection against theft, diversion, or sabotage.

D. Other Events (i.e., concerning design, analysis, construction, testing, operation, use or disposal of licensed facilities or regulated materials);

1. An accidental criticality (10 CFR 70.52 (a)).

(2) A major deficiency in design, construction or operation having safety implication requiring immediate remedial action.

3. Serious deficiency in management or procedural controls in major areas.

4. Series of events (where individual events are not of major importance), recurring incidents, and incidents with implications for similar facilities (generic incidents), which create major safety concern.

II. *For Commercial Nuclear Power Plants:*

A. Malfunction of Facilities, Structures or Equipment:

1. Exceeding a safety limit of license Technical Specifications (10 CFR 50.36(c)).

2. Major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary.

3. Loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

B. Design or Safety Analysis Deficiency, Personnel Error or Procedural or Administrative Inadequacy:

1. Discovery of a major condition not specifically considered in the Safety Analysis Report (SAR) or technical specifications that require immediate remedial action.

2. Personnel error or procedural deficiencies which result in loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

III. *For Fuel Cycle Licensees:*

A. For Reprocessing Facilities:

1. A safety limit of license Technical Specifications is exceeded and a plant shutdown is required (10 CFR 50.36 (c)).

2. A major condition not specifically considered in the Safety Analysis Report or technical specifications that required immediate remedial action.

B. All Fuel Licensees:

1. An event which seriously compromised the ability of a confinement system to perform its designated function.

Effective date: This general statement of policy shall be effective February 24, 1977.

Dated at Washington, D.C. this 23rd day of February, 1977.

For the Nuclear Regulatory Commission.

SAMUEL J. CHILK,
Secretary of the Commission.

POOR ORIGINAL

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[2] Information relating to certain incidents may either be classified or under consideration for classification because of national security implications. Classified information will be withheld when formally reporting these incidents per Section 208. Any classified details regarding such incidents would be available to the Congress, upon request, under appropriate security arrangements.

[3] These examples are not all inclusive. Other incidents of similar significance will be considered for reporting.

[4] Information relating to certain incidents may either be classified or under consideration for classification because of national security implications. Classified information will be withheld when formally reporting these incidents per Section 208. Any classified details regarding such incidents would be available to the Congress, upon request, under appropriate security arrangements.

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REPORT TO CONGRESS
ON
ABNORMAL OCCURRENCES

October - December 1978

Status as of January 31, 1979
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Office of Management and Program Analysis
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report, the fifteenth in the series, covers the period from October 1 to December 31, 1978.

The following incidents or events, including those submitted by the Agreement States, in that time period were determined by the Commission to be significant and reportable:

1. There was one abnormal occurrence at the 70 nuclear power plants licensed to operate. The event involved a loss of containment integrity at two pressurized water reactors.

2. There were no abnormal occurrences at fuel cycle facilities (other than nuclear power plants).

3. There were no abnormal occurrences at other licensee facilities.

4. There were two abnormal occurrences reported by the Agreement States. One event involved a radiation overexposure of a radiographer's assistant. The other involved transportation of a package with radiation levels in excess of limits.

This report also contains information updating previously reported abnormal occurrences.

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PREFACE

INTRODUCTION

The Nuclear Regulatory Commission reports to the Congress each quarter under provisions of Section 208 of the Energy Reorganization Act of 1974 on any abnormal occurrences involving facilities and activities regulated by the NRC. An abnormal occurrence is defined in Section 208 as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety.

Events are currently identified as abnormal occurrences for this report by the NRC using the criteria delineated in Appendix A. These criteria were promulgated in an NRC policy statement which was published in the Federal Register (42 FR 10950) on February 24, 1977. In order to provide wide dissemination of information to the public, a Federal Register notice is issued on each abnormal occurrence with copies distributed to the NRC Public Document Room and all local public document rooms. At a minimum, each such notice contains the date and place of the occurrence and describes its nature and probable consequences.

The NRC has reviewed Licensee Event Reports, licensing and enforcement action (e.g., violations, infractions, deficiencies, civil penalties, license modifications, etc.), generic issues, significant inventory differences involving special nuclear material, and other categories of information available to the NRC. The NRC has determined that only those events, including those submitted by the Agreement States, described in this report meet the criteria for abnormal occurrence reporting. This report, the fifteenth in the series, covers the period between October 1 December 31, 1978. Events which occurred during this quarter and are later determined to be abnormal occurrences will be included in the next quarterly report. Some events require considerable time and effort to analyze due to the complexity of situations where actual consequences are not readily apparent and additional facts are required.

Information reported on each event includes: date and place; nature and probable consequences; cause or causes; and actions taken to prevent recurrence.

THE REGULATORY SYSTEM

The system of licensing and regulation by which NRC carries out its responsibilities is implemented through rules and regulations in Title 10 of the Code of Federal Regulations. To accomplish its objectives, NRC regularly conducts licensing proceedings, inspection and enforcement activities, evaluation of operating experience and confirmatory research, while maintaining programs for establishing standards and issuing technical reviews and studies. The NRC's role in regulating represents a complete cycle, with the NRC establishing standards and rules; issuing licenses and permits; inspecting for compliance; enforcing license requirements; and carrying on continuing evaluations, studies and research projects to improve both the regulatory process and the protection of the public health and safety. Public participation is an element of the regulatory process.

In the licensing and regulation of nuclear power plants, the NRC follows the philosophy that the health and safety of the public are best assured through the establishment of multiple levels of protection. These multiple levels can be achieved and maintained through regulations which specify requirements which will assure the safe use of nuclear materials. The regulations include design and quality assurance criteria appropriate for the various activities licensed by NRC. An inspection and enforcement program helps assure compliance with the regulations. Stringent requirements for reporting incidents or events exist which help identify deficiencies early enough to prevent serious consequences and aid in assuring that prompt and effective corrective action is taken to prevent their recurrence.

Most NRC licensee employees who work with radioactive materials are required to utilize personnel monitoring devices such as film badges or TLD (thermoluminescent dosimeter) badges. These badges are processed periodically and the exposure results normally serve as the official and legal record of the extent of personnel exposure to radiation during the period the badge was worn. If an individual's past exposure history is known and has been sufficiently low, NRC regulations permit an individual in a restricted area to receive up to three rems of whole body exposure in a calendar quarter. Higher values are permitted to the extremities or skin of the whole body. For unrestricted areas, permissible levels of radiation are considerably smaller. Permissible doses for restricted areas and unrestricted areas are stated in 10 CFR Part 20. In any case, the NRC's policy is to maintain radiation exposures to levels as low as reasonably achievable.

REPORTABLE OCCURRENCES

Since the NRC is responsible for assuring that regulated nuclear activities are conducted safely, the nuclear industry is required to report incidents or events which involve a variance from the regulations, such as personnel overexposures, radioactive material releases above prescribed limits, and malfunctions of safety-related equipment. Thus, a reportable occurrence is any incident or event occurring at a licensed facility or related to licensed activities which NRC licensees are required to report to the NRC. The NRC evaluates each reportable occurrence to determine the safety implications involved.

Because of the broad scope of regulation and the conservative attitude toward safety, there are a large number of events reported to the NRC. The information provided in these reports is used in the NRC and the industry in their continuing evaluation and improvement of nuclear safety. Most of the reports received from licensed nuclear power facilities describe events that did not directly involve the nuclear reactor itself, but involved equipment and components which are peripheral aspects of the nuclear steam supply system, and are minor in nature with respect to impact on public health and safety. The majority are discovered during routine inspection and surveillance testing and are corrected upon discovery. Typically, they concern single malfunctions of components or parts of systems, with redundant operable components or systems continuing to be available to perform the design function.

Information concerning reportable occurrences at facilities licensed or otherwise regulated by the NRC is routinely disseminated by NRC to the nuclear industry, the public, and other interested groups as these events occur. Dissemination includes deposit of incident reports in the NRC's public document rooms, special notifications to licensees and other affected or interested groups, and public announcements. In addition, a biweekly computer printout containing information on reportable events received from NRC licensees is sent to the NRC's more than 120 local public document rooms throughout the United States and to the NRC Public Document Room in Washington, D.C.

The Congress is routinely kept informed of reportable events occurring at licensed facilities.

AGREEMENT STATES

Section 274 of the Atomic Energy Act, as amended, authorizes the Commission to enter into agreements with States whereby the Commission relinquishes and the States assume regulatory authority over byproduct, source and special nuclear materials (in quantities not capable of sustaining a chain reaction). Compatible and compatible programs are the basis for agreements.

Presently, information on reportable occurrences in Agreement State licensed activities is publicly available at the State level. Certain information is also provided to the NRC under exchange of information provisions in the agreements. NRC prepares a semiannual summary of this and other information in a document entitled, "Licensing Statistics and Other Data," which is publicly available.

In early 1977 the Commission determined that abnormal occurrences happening at facilities of Agreement State licensees should be included in the quarterly report to Congress. The abnormal occurrence criteria included in Appendix A is applied uniformly to events at NRC and Agreement State licensee facilities. Procedures have been developed and implemented and any abnormal occurrences reported by the Agreement States to the NRC are included in these quarterly reports to Congress.

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

OCTOBER-DECEMBER 1978

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the 70 nuclear power plants licensed to operate during the fourth quarter of 1978. Through the end of December, the NRC had determined that the following event was an abnormal occurrence.

78-5 Loss of Containment Integrity

Preliminary information pertaining to this incident was reported in the *Federal Register* (43 FR 60350). Appendix A (Example 2 of "For Commercial Nuclear Power Plants") of this report notes that a major degradation of the primary containment boundary can be considered an abnormal occurrence.

Date and Place - On July 26, 1978, the Northeast Nuclear Energy Company (NNECO) reported to the NRC an event at Millstone Unit 2, a pressurized water nuclear plant located in New London County, Connecticut. On September 8, 1978, the Public Service Electric and Gas Company (PSE&G) reported a similar event at Salem Unit 1, a pressurized water nuclear plant located in Salem County, New Jersey.

Nature and Probable Consequences - The events reported at Millstone Unit 2 and Salem Unit 1 involved loss of automatic valve closure capability for certain large sized isolation valves in the containment ventilation systems while the valves were open for containment purging operations. Such loss of closure capability significantly degraded the containment leakage retention integrity for extended time periods (hours to days) in some cases while the units were operating at power. The automatic closure feature was lost because the signals which are intended to initiate automatic closure under certain accident conditions were either bypassed or overridden and therefore ineffective. Normal purging activities do not require the negating of any automatic closure signals.

No radiological accidents occurred during these periods and therefore these safety features were not challenged. However, if a design basis Loss of Coolant Accident (LOCA) had occurred under these conditions, the offsite consequences would have increased above those anticipated with the automatic containment isolation valve closure feature operable. In addition, as described below, the performance of the Emergency Core Cooling Systems (ECCS) may have also been degraded due to a decrease in the pressure buildup inside the containment during the accident. The containment systems and the emergency core cooling systems (Figure 1) are two of many safety features at nuclear power plants.

PRESSURIZED WATER REACTOR

CONTAINMENT VENTILATION SYSTEMS

(SIMPLIFIED DIAGRAM)

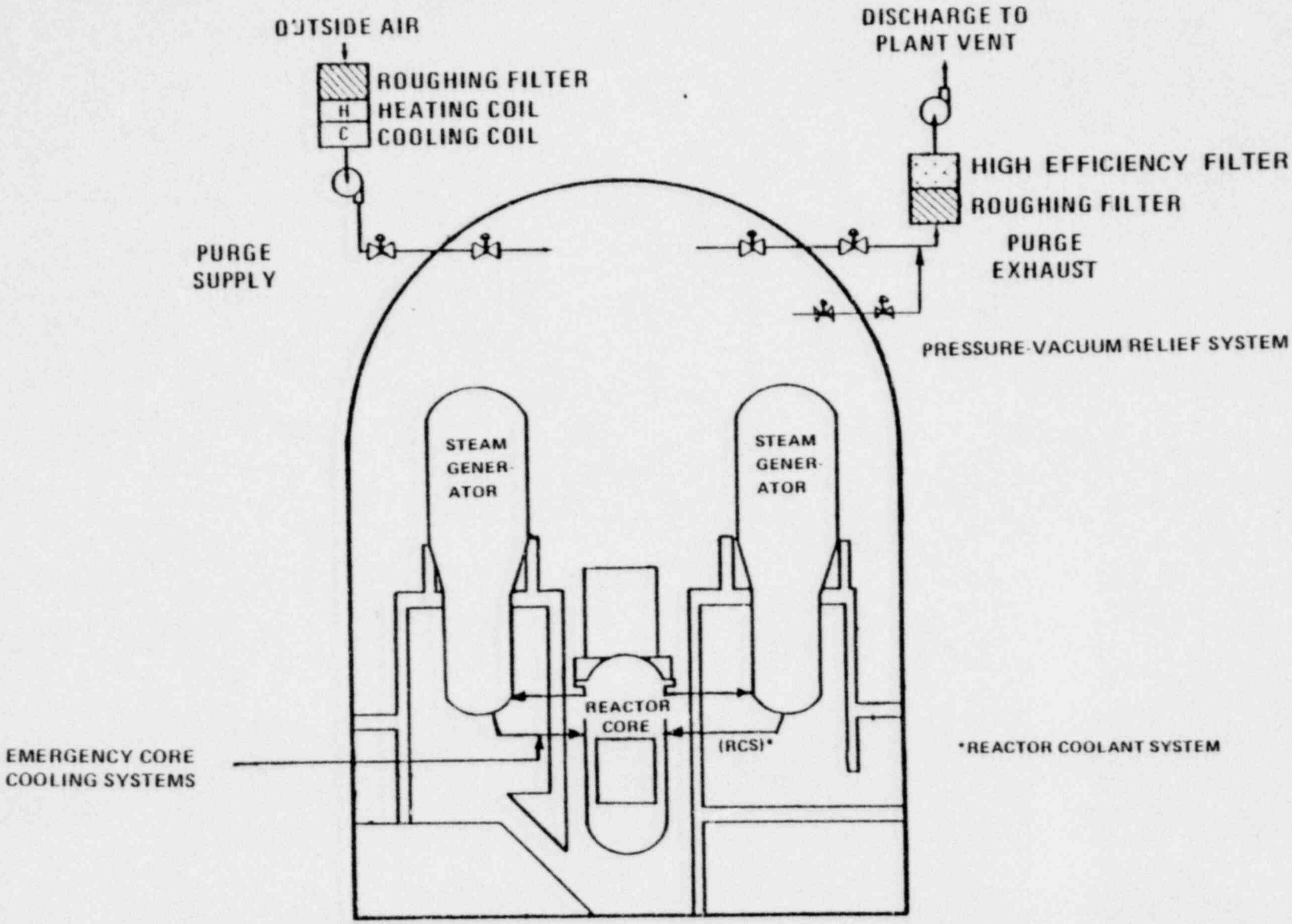

FIGURE 1

The containment systems include a large thick-walled building or vessel surrounding the reactor and its primary coolant system which is designed to be a leak tight enclosure to limit accidental releases of radioactivity to the environment. The emergency core cooling systems (ECCS) are designed to automatically supply water to the reactor core to mitigate the consequences of postulated accidents where the normal cooling water is lost.

Containment buildings are designed with ventilation or purging systems which can add fresh air and exhaust the containment atmosphere to maintain the temperature, humidity, pressure and the radioactivity levels within acceptable limits. (As shown in Figure 1, the exhausted air is normally filtered to reduce the amount of radioactivity and particles released to the environment. It is also monitored to prevent releases in excess of the limits in the Technical Specifications.) These systems must be isolated to provide for containment integrity, when needed. For this purpose, each purge inlet and outlet pipe has two isolation valves, as do the other containment penetrations. These valves, which are permitted to be opened for purging during normal plant operation, are designed to automatically close in a very short time period, if needed for containment isolation. At many of the operating nuclear power plants, the normal containment purge systems use large pipes--ranging from two feet to more than five feet in diameter.

Purging does not occur continually at these plants (i.e., the containment isolation valves are normally closed). As mentioned earlier, when containment isolation valves are opened for purging, the automatic valve closure signals are normally operable. A containment isolation actuation signal, such as the containment high radiation signal, would then initiate rapid closing of the valves well within the valve closure times included in the design basis assumptions. The offsite consequences of any radioactive releases to the environs would be as realistically evaluated in the Environmental Impact Statement. However, in the unlikely event of a postulated design basis loss-of-coolant accident (LOCA) while purging with these containment valves open and their closure signals inoperative, the containment would have an unacceptably high leakage rate of radioactivity to the environment. This increased containment leakage rate would also result in a reduced containment pressure buildup during the postulated accident which, in turn, could degrade the ECCS performance. Parametric calculations indicate that the low containment pressure would result in a calculated reduction in core reflood rate and heat removal capability and would lead to higher calculated fuel cladding temperatures.

If the LOCA occurred while purging with the valve closure signals negated, although operator action may result in a decrease in the radioactive release to the environs, it is unlikely that operator action could be taken in time to close the valves to prevent degraded ECCS performance since (1) blowdown would be complete and peak containment pressure would be reached in approximately one to two minutes, and (2) there would be no indication to the operator that the valve closure signals had been bypassed. The only means the operator would have had to identify the valve status would be the valve position indication. Even if the operator were to take manual action, it is questionable whether the valves would be able to close against the postulated blowdown flow rate through the penetrations unless this action is taken within a few seconds following the accident.

The details of the two reported events were:

Millstone Unit 2 Event - During a review of operating procedures on July 25, 1978, the licensee discovered that since May 1, 1978, intermittent containment purge operations had been conducted with the isolation signals to the redundant containment isolation valves in the purge inlet and outlet (48 inch butterfly valves) manually overridden and inoperable. The isolation signals were manually overridden to purge the containment with a "high radiation" signal present--see further details below. (This "high radiation" signal was actually a low value set to initiate actions in a conservative manner.) The manual override circuitry not only defeated the "high radiation" actuation signal to close these valves, but also bypassed all other isolation signals to these valves.[1] The operator had no indication that this bypass condition existed and, consequently, was not aware that operator action would be required to close the valves in the event of an accident.

From May 1 to July 25, 1978 (about 2,000 hours), the containment of Millstone Unit 2 was purged to reduce radioactivity levels, for interim periods ranging from 5 minutes to 31 hours, with purging occurring for approximately 9% (180 hours total) of the total time period. For each purge, the levels of radioactivity released to the environment were monitored and were within Technical Specification requirements. However, the regulations and specifically the plant Technical Specifications both require the containment isolation valves in lines that open directly to the containment atmosphere be capable of automatic closure during purging or other operations, or such lines must be acceptable on some other defined basis to mitigate the potential consequences of postulated design basis accidents.

Salem Unit 1 Event - On September 8, 1978, the NRC was advised that, as a matter of routine, Salem Unit 1 has been "venting" the containment through the containment ventilation system valves to reduce pressure. In certain

[1]To manually override a safety actuation signal, the operator cycles the valve control switch to the closed position and then to the open position. This action energizes a relay which overrides the safety signal and allows manual operation independent of any safety actuation signal. This circuitry is designed in this manner to permit reopening of certain valves after an accident to allow manual operation of safety equipment.

instances, this venting has occurred with the containment "high particulate" radiation monitor isolation signal overridden to the purge valves (36-inch diameter valves) and pressure-vacuum relief vent valves (10-inch diameter valves). Override of the containment isolation signal was accomplished by resetting the train A and B reset buttons.[2] Under these circumstances, six valves in the containment vent and purge systems could be opened with a high particulate isolation signal present. This override was performed after verifying the actual containment particulate radiation levels were acceptable for venting. The licensee, after further investigation of this practice, determined that the reset of the particulate alarm also bypasses the containment isolation signal to the purge valves and the vent valves and, therefore, these valves would also not have automatically closed as required in the event of a signal to initiate emergency core cooling.

The licensee had modified its procedure to preclude venting of the containment through the purge valves when the containment "high particulate" alarm exists.

Cause or Causes - The events resulted from procedural inadequacies and design deficiencies. While the containment atmospheres were properly sampled and the purging (venting) discharges that actually occurred were within regulatory requirements, the procedures did not adequately address the operability of the isolation valves and the limitations on negating the closure signals. The requirements for having the valves capable of closing automatically were not discussed, nor were the related Technical Specifications referenced, in the procedures. Although not a requirement, to do so is good practice. Design deficiencies contributed to the event in that (1) bypassing one safety signal also bypassed other safety signals, and (2) the use of this bypass was not annunciated in the control room.

Action Taken to Prevent Recurrence

Licensees

Northeast Nuclear Energy Company (NNECO) - The immediate corrective action taken by NNECO, at Millstone Unit 2, was to close, deenergize and administratively remove the containment purge valves from service (tag out). Future NNECO actions include the development of procedure revisions and submission of proposed changes to the Millstone Unit 2 Technical Specifications. These changes would allow somewhat higher containment radiation monitor setpoints, still based on remaining well within allowable effluent release limits, which will permit containment purging over a wide range of normal containment conditions without overriding the "high radiation" signal or any other signal

[2]These buttons reset the logic circuits (train A and train B) associated with containment isolation.

<u>Public Service Electric and Gas Company (PSE&G)</u> - The immediate corrective action taken by PSE&G at Salem Unit 1, was to stop venting when override of the valve closure signals is involved. In addition, PSE&G initiated design changes such that the isolation valves will be closed automatically by a signal actuating the emergency core cooling systems even if the containment radiation monitor alarms are overridden.

<u>NRC</u> - The NRC has previously reviewed the practice of purging containment during operation and, in 1975, revised its licensing program for new plants by restricting purging during operation.

In a related instance in May 1976, Commonwealth Edison Company reported that they had stopped the practice of purging containment during operation at Zion Station after having determined that a safety analysis had not been performed to assess the affect of purge valve closure time upon ECCS performance. Preliminary calculations by Commonwealth Edison Company indicated that containment pressure would fall below that assumed for the design basis loss-of-coolant accident and that the existing analysis was therefore not conservative. The NRC review of this event concluded that the effect of purging upon ECCS performance was generic to many operating reactors. However, the significance of this event for public health and safety was determined to be minor when evaluated on more realistic bases, i.e., automatic valve closure times on the order of 5 to 10 seconds, with peak pressure in containment being reached on the order of 1-2 minutes. This item was therefore given relatively low priority for resolution.

Now based on an assessment of these events, the NRC believes that tighter controls are warranted on purging and venting operations to assure containment integrity at these two and other nuclear power plants.

In addition to reviewing the licensees' corrective actions for these events, the NRC staff is reviewing the generic implications for other facilities.

On November 29, 1978, a generic letter was sent by the NRC to all operating reactor licensees which requested the licensees to commit to stop purging during operation or to provide a basis why purging during operation should be permitted. The NRC basis for allowing limited purging through fast closing valves during operation will be a demonstration of the capability of the valves to close under postulated accident conditions and a Technical Specification limitation of 90 hours per year for purging during operation. The NRC basis for allowing unlimited purging through fast closing valves during operation will require: (1) demonstration of the capability of the valves to close under postulated accident conditions, (2) an assessment demonstrating the acceptability of purging during operation upon emergency core cooling system performance, (3) containment purge and isolation instrumentation, and control circuit designs which conform to the appropriate safety standards (IEEE Standard 279-1971), and

(4) an assessment demonstrating acceptability of the radiological consequences of the design basis loss-of-coolant accident initiated during purge operations.

Also, the licensees were requested to review all safety-actuation signal circuit designs which incorporate a manual override, or negating, feature to insure (1) that override of a single safety actuation signal does not bypass other safety actuation signals, and (2) that the use of the manual override feature is appropriately annunciated in the control room. Licensees will be required to report the results of their review and their corrective actions for any nonconforming circuits.

In addition, the licensees were advised of the necessity for proper management controls for the use of manual override of safety signals during nonemergency conditions. NRC will, through their inspection program, assure that licensees have initiated appropriate follow-up action.

On December 29, 1978, an NRC Inspection & Enforcement Circular was sent to all Construction Permit holders which addressed the NRC concerns for unintentional bypass of isolation and safety actuation signals. The Circular did not address containment purging since this issue is specifically addressed in the Standard Review Plan (SRP) and was only of concern for those operating reactors not reviewed against SRP Section 6.2.4.

Future reports will be made as appropriate.

FUEL CYCLE FACILITIES

(Other Than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the fourth quarter of 1978. Through the end of December, the NRC had not determined that any events were abnormal occurrences.

OTHER NRC LICENSEES

(Industrial Radiographers, Medical Institutions, Industrial Users, etc.)

There are currently more than 8,000 NRC nuclear material licenses in effect in the United States, principally for use of radioisotopes in the medical, industrial and academic fields. Incidents were reported in this category from licensees such as radiographers, medical institutions, and byproduct material users.

The NRC is reviewing events reported by these licensees during the fourth quarter of 1978. Through the end of December, the NRC had not determined that any events were abnormal occurrences.

AGREEMENT STATE LICENSEES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (see Appendix A) and report the events to the NRC for inclusion in this report. During the fourth quarter of 1978, the following Agreement State licensee event was determined reportable as an abnormal occurrence.

AS78-5 Overexposure of a Radiographer's Assistant

Appendix A (Example 1 of "For All Licensees") of this report notes that an exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation can be considered an abnormal occurrence.

Date and Place - On November 4, 1978, Pittsburgh Testing Laboratory notified the Louisiana Nuclear Energy Division of an overexposure to a radiographer's assistant at the Freeport Chemical Company plant at Uncle Sam, Louisiana. The incident occurred on November 4, 1978 while radiographing storage tanks.

Nature and Probable Consequences Following a radiographic exposure the radiographer's assistant approached a Gamma Century exposure device which was on some scaffolding approximately 30 feet above ground level inside a storage tank. He locked the exposure device without surveying it and then proceeded to remove the source guide tube, at which time he noticed the capsule was protruding from the outlet nipple approximately 1/4-inch. Thinking that the capsule may be loose, he replaced the guide tube and notified the radiographer in charge. The radiographer found that a 3/4-turn of the crank handle after unlocking the exposure device returned the source to the shielded position. The assistant's and the radiographer's dosimeters were off-scale, and both film badges were returned for immediate processing. The film badge supplier reported a whole-body dose of 3.5 rems to the radiographer's assistant and 410 millirem to the radiographer. From a re-enactment of the incident, it was calculated that the radiographer's assistant received a dose of from 800 to 1000 rems to his right hand. No hand exposure was received by the radiographer.

No outward evidence of injury to the assistant radiographer's hand has occurred. Shortly after the incident, the employee noticed some discomfort in his hand and consulted a physician; however, it was diagnosed as tendinitis, rather than due to radiation injury.

Cause or Causes - The primary cause of this excessive exposure was the failure of the radiographer's assistant to perform a survey of the exposure device to insure that the source had been returned to the shielded position. A contributing factor in the incident was the assistant's false impression that if the device is locked, the source is in the safe position.

Actions Taken to Prevent Recurrence

Licensee - The licensee has restricted the radiographer's assistant's work for the remainder of the calendar quarter and has reinstructed all personnel in the proper use of survey meters.

Louisiana Nuclear Energy Division - The Louisiana Nuclear Energy Division has cited the licensee with appropriate violations for the excessive exposure and the failure to make a proper survey.

This incident is closed for purposes of this report.

AS78-6 Transportation of Package with Radiation Levels in Excess of Limits

Appendix A (Example 2 of "For All Licensees") of this report notes that an exposure to an individual in an unrestricted area such that the whole body dose received exceeds 0.5 rem in one calendar year can be considered an abnormal occurrence.

The incident involved members of the public and licensee personnel in both Agreement and non-Agreement States. The shipment originated in Colorado, an Agreement State, and all estimated personnel exposures in excess of the Abnormal Occurrence threshold occurred in Colorado.

Date and Place - On October 18, 1978, the NRC was notified by one of its licensees, Technical Operations, Inc., of Burlington, Massachusetts, that a package containing a radioactive source had been received with external radiation levels exceeding NRC and DOT regulations. The package had been shipped to Technical Operations by Testing Consultants, Inc., a Colorado licensee. Testing Consultants notified Colorado radiation control personnel of the incident on October 18, 1978.

Nature and Probable Consequences - A package containing an industrial radiographic source was shipped from Denver, Colorado to Burlington, Massachusetts through airports in Memphis, Tennessee, and Boston, Massachusetts. Upon receipt at its destination, the package was being carried into the Technical Operations, Inc. receiving area when a nearby radiation monitor alarmed. A radiation survey showed radiation levels of 250 millirem per hour at one meter from the top of the package, considerably more than expected. Based on this measured dose rate, radiation levels at the surface of the package were calculated to be up to 10 rem per hour. Subsequent investigations by NRC inspectors and representatives of the States of Colorado, Tennessee, and Massachusetts showed that up to 32 people had handled or been in proximity to the package at sometime during shipment. The calculated radiation exposure to most of these individuals was very small; however, it is estimated that four individuals may have received radiation exposures between 500 and 720 millirem. The actual radiation exposures received by these individuals are likely to have been small, since worst case circumstances were used in estimating the radiation exposures. These radiation exposures are well below radiation exposures necessary for clinical manifestations of radiation injury. However, they do exceed the abnormal occurrence reporting threshold of 500 millirem exposure in one calendar year to persons in an unrestricted area.

On October 16, 1978, a Technical Operations, Inc., Model 750 Radiographic Source Changer was loaded with a decayed nominal 11 curie iridium-192 sealed source by Testing Consultants, Inc., a State of Colorado licensee for return to Technical Operations, Inc., the source supplier. The source changer is used to transport new and decayed sealed sources used in radiographic devices during the performance of industrial radiography. The sealed source is attached to a long teleflex drive cable which is used to vary the position of the source from a locked (fully shielded) position to positions used for radiography when the source is in the radiographic device. Subsequent to transfer from the radiographic device to the source changer, the radioactive source was locked in place in the changer, and a radiation survey was made which showed expected low radiation levels. The teleflex drive cable was then coiled to prepare the changer for shipment. Difficulty was experienced in coiling the cable. Due to the screw-like nature of the teleflex drive cable, rotation of the cable most likely caused the source to move from the fully shielded position at this time. No additional surveys were made of the changer prior to shipment. During the course of shipment, 32 people handled or were in proximity to the changer, including a secretary, 4 truck drivers, 5 cargo handlers, 5 flight personnel on 2 cargo aircraft, and 17 airport ground personnel. All exposures estimated at 500 millirem or more occurred in Colorado. From calculations it is estimated that a secretary at Testing Consultants received 720 millirem (the package was near the secretary's desk for about six hours), two Federal Express cargo handlers received 500 millirem each, and a Federal Express hazardous materials handler received 540 millirem.

Cause or Causes - Based on a demonstration by Technical Operations personnel, NRC inspectors concluded that it is possible to lock a Technical Operations Model 750 Source Changer without the radioactive source being in a fully shielded position and that it is possible for the radioactive source to move from the fully shielded position if the teleflex drive cable is rotated when the teleflex cable is coiled for shipment. A radiation survey, made after the package was completely ready for shipment, would have detected either circumstance. Such a survey was required by Testing Consultants procedures, but was not performed by Testing Consultants.

Action Taken to Prevent Recurrence

State of Colorado As a result of the State's investigation, Testing Consultants was cited for five items of noncompliance. The two most significant items were (1) failure to perform and record a survey of the loaded shipping container, and (2) allowing radiation levels in unrestricted areas which could result in an individual receiving a dose in excess of 2 millirems in any one hour (or 500 millirems in any calendar year).

Testing Consultants The licensee stated that individuals would be instructed to perform a radiation survey of each shipping container immediately prior to delivering the package to a carrier for transport.

Technical Operations Technical Operations personnel indicated they plan to modify the operating instructions for the Model 750 Source Changer to call special attention to this occurrence and stress the importance of a radiation survey after the changer is completely ready for shipment. Technical Operations personnel dissassembled the changer locks and examined them for wear. It was found that all components met original engineering specifications. Technical Operations is also reviewing the possibility of a design change to prevent the source from moving if the teleflex drive cable is rotated.

NRC and State Representatives - The circumstances surrounding this incident were thoroughly investigated by NRC inspectors and State representatives. The calculated exposures are based on a detailed reenactment of the handling of the package from its preparation for shipping to its destination.

This incident is closed for purposes of this report.

APPENDIX A

ABNORMAL OCCURRENCE CRITERIA

The following criteria for this report's abnormal occurrence determinations were set forth in an NRC policy statement published in the Federal Register (42 FR 10950) on February 24, 1977.

Events involving a major reduction in the degree of protection of the public health or safety. Such an event would involve a moderate or more severe impact on the public health or safety and could include but need not be limited to:

1. Moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the Commission;

2. Major degradation of essential safety-related equipment; or

3. Major deficiencies in design, construction, use of, or management controls for licensed facilities or material.

Examples of the types of events that are evaluated in detail using these criteria are:

For All Licensees

1. Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual to 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation (10 CFR Part 20.403(a)(1)), or equivalent exposures from internal sources.

2. An exposure to an individual in an unrestricted area such that the whole body dose received exceeds 0.5 rem in one calendar year (10 CFR Part 20.105(a)).

3. The release of radioactive material to an unrestricted area in concentrations which, if averaged over a period of 24 hours, exceed 500 times the regulatory limit of Appendix B, Table II, 10 CFR Part 20 (10 CFR Part 20.403(b)).

4. Radiation or contamination levels in excess of design values on packages, or loss of confinement of radioactive material such as: (a) a radiation dose rate of 1,000 mrem per hour three feet from the surface of a package containing the radioactive

material, or (b) release of radioactive material from a package in amounts greater than the regulatory limit (10 CFR Part 71.36(a)).

5. Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas.

6. A substantiated case of actual or attempted theft or diversion of licensed material or sabotage of a facility.

7 Any substantiated loss of special nuclear material or any substantiated inventory discrepancy which is judged to be significant relative to normally expected performance and which is judged to be caused by theft or diversion or by substantial breakdown of the accountability system.

8. Any substantial breakdown of physical security or material control (i.e., access control, containment, or accountability systems) that significantly weakened the protection against theft, diversion or sabotage.

9. An accidental criticality (10 CFR Part 70.52(a)).

10. A major deficiency in design, construction or operation having safety implications requiring immediate remedial action.

11. Serious deficiency in management or procedural controls in major areas.

12. Series of events (where individual events are not of major importance), recurring incidents, and incidents with implications for similar facilities (generic incidents), which create major safety concern.

For Commercial Nuclear Power Plants

1. Exceeding a safety limit of license Technical Specifications (10 CFR Part 50.36(c)).

2. Major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary.

3. Loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

4. Discovery of a major condition not specifically considered in the Safety Analysis Report (SAR) or Technical Specifications that require immediate remedial action.

5. Personnel error or procedural deficiencies which result in loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod systems)

For Fuel Cycle Licensees

1. A safety limit of license Technical Specifications is exceeded and a plant shutdown is required (10 CFR Part 50.36(c))

2. A major condition not specifically considered in the Safety Analysis Report or Technical Specifications that requires immediate remedial action.

3. An event which seriously compromised the ability of a confinement system to perform its designated function.

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the October through December 1978 period, the NRC, NRC licensees and other involved parties, such as reactor vendors and architects and engineers, continued with the implementation of actions necessary to prevent recurrence of previously reported abnormal occurrences. The referenced Congressional abnormal occurrence reports below provide the initial and any updating information on these abnormal occurrences. Those occurrences not now considered closed will be discussed in subsequent reports in the series.

NUCLEAR POWER PLANTS

The following abnormal occurrence was originally reported in NUREG-75/090, "Report to Congress on Abnormal Occurrences: January-June 1975," and updated in subsequent reports in this series, i.e., NUREG-0090-1 and 6. It is further updated as follows:

75-7 <u>Steam Generator Feedwater Flow Instability at Pressurized Water Reactors</u>

Since the previous 1977 update of this item (NUREG-0090-6), additional incidents of steam generator water hammer have occurred at pressurized water reactors (PWRs). Between January 1, 1977 and December 31, 1978, five water hammer events were reported in PWR steam generator feedwater systems. A related event also occurred in a feedwater system which was due to dynamic instability of the feedwater regulating valves. In all events reported to the NRC, the damage was minor and did not pose a threat to the health and safety of the public. Damage occurred in only four of the above events and was limited to two snubbers, two cracked welds, and three feedwater regulating valves.

Steam generator water hammer has occurred in certain nuclear power plants as a result of the rapid condensation of steam in a steam generator feedwater line. The consequent acceleration of a slug of water which upon impact ("hammering") within the piping system causes undue stresses in the piping and its support system. The significance of these events varies from plant to plant. Since the total loss of feedwater could affect the ability of the plant to cool down after a reactor shutdown, the NRC is concerned about these events occurring, even though an event with potentially serious consequences is unlikely to happen.

Because of the continuing occurrence of water hammer events, the NRC, in September 1977, informed all PWR licensees that water hammer events due to the rapid condensation of steam in the feedwater lines of steam generators represented a safety concern and that further actions by licensees for Westinghouse and Combustion Engineering designed nuclear steam supply systems are warranted to assure that an acceptably low risk to public safety due to such events is maintained. Accordingly, these licensees were requested to submit proposed hardware and/or procedural modifications, if any, which would be necessary to assure that the feedwater lines and feedrings remain filled with water during normal as well as transient operating conditions. At the same time, the NRC provided each PWR licensee with a copy of its consultant's report, "An Evaluation of PWR Steam Generator Water Hammer," NUREG-0291. All 22 licensed operating PWR facilities responded to the NRC September 1977 letter. The NRC has completed review of five of these and concluded that the proposed modifications to reduce the occurrence of water hammer are acceptable. The NRC expects to complete the review of the remaining 17 facilities in early fiscal year 1980. The means employed to reduce water hammer includes the installation of loop seals, J-tubes to the feedwater sparger, and automatic initiation of the auxiliary feedwater flow to maintain the sparger and associated piping full of water to preclude water hammer. Additionally, administrative controls have been established to limit the flow of auxiliary feedwater when a feedwater line might contain steam.

The previously reported incidents of feedwater instability that occurred at the Duquesne Light Company's Beaver Valley Unit 1 in Beaver County, Pennsylvania have been analyzed and corrective action has been taken. These events involved the interaction of water hammer pressure waves with a feedwater control valve that caused sustained oscillations in valve motion and pressure. The balanced double plugs in all feedwater control valves were replaced with ported cylinders to correct this type of instability.

As mentioned in the earlier reports, design margins have been adequate to limit the consequences of these occurrences. Also, redundant means do exist for plant cool down after a reactor shutdown should a loss of feedwater occur to the steam generators.

Future reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090-5, "Report to Congress on Abnormal Occurrences: July-September 1976," and updated in a subsequent report in this series, i.e., NUREG-0090-8. It is further updated as follows:

76-11 <u>Steam Generator Tube Integrity</u>

Since the 1977 update of the item (NUREG-0090-8), the following significant developments related to pressurized water reactor (PWR) steam generator tube integrity have occurred.

Westinghouse Designed Units

Degradation of steam generator tubes, due to a corrosion-related phenomenon known as "denting," has continued in Westinghouse manufactured pressurized water reactor systems. Affected units include Surry Units 1 and 2, Turkey Point Units 3 and 4, Connecticut Yankee, R. E. Ginna, Indian Point Units 2 and 3, Point Beach Units 1 and 2, H. B. Robinson, and San Onofre.

Steam generator replacement programs at Surry Units 1 and 2 have been reviewed and the Safety Evaluation Report published. Replacement of Surry Unit 2 steam generators is tentatively scheduled to begin in early February of 1979 and Unit 1 steam generators in the fall of 1979. Replacement programs for Turkey Point Units 3 and 4 are under review.

Yankee Rowe and Point Beach Unit 1 experienced small steam generator tube leaks due to a phenomenon designated as "deep crevice cracking."

Combustion Engineering (CE) Designed Units

Degradation of steam generator tubes due to "denting" has continued at Maine Yankee and Millstone Unit 2. Modifications at these and other similar units have been made to mitigate the effects of "denting."

Consumers Power Company has stated that they are considering replacement of the Palisades steam generators because over 20 percent of the Palisades steam generator tubes have been plugged due to an earlier tube wastage problem which has generally been arrested.

During an April 1978 inspection, minor denting was discovered in the St. Lucie Unit 1 steam generators. A program for chemical cleaning is under review by the NRC and cleaning is planned for April 1979.

Babcock & Wilcox (B&W) Designed Units

Between July 1977 and April 1978, two tube leaks occurred at Oconee Unit 1, and one tube leak at Oconee Unit 2, and one tube leak at Oconee Unit 3. Three of these leaks were due to circumferential fatigue cracks and one leak in Unit 1 was a leaking tube weld. No leaks have occurred since April 1978.

Duke Power Company and B&W have undertaken a tube sleeving demonstration program at Oconee Unit 1. The purpose of the tube sleeves is to reduce the dynamic stresses in the region of previously indicated tube abnormalities.

During the October 1978 steam generator tube inspection and tube plugging operations at Oconee Unit 1, two steam generator tube plugs were lost in the primary coolant system. Further details are provided below.

Details of the experiences by the three PWR reactor designers since mid-1977 are described below.

Westinghouse

Steam generator tube "denting" is a corrosion-related phenomenon resulting from the build-up of support plate corrosion product in the annulus between the tubes and the tube support plates. In-plane forces caused by the build-up of corrosion products eventually cause "denting" of the tubes and deformation of the tube support plates. This phenomenon has resulted in stress corrosion cracking and leaks at the tube/tube support plate intersections and in the U-bend sections of tubes which were highly stressed because of support plate deformation. Denting has been observed in plants which made a change over from phosphate to all-volatile secondary water chemistry treatments and in plants which have operated exclusively with the all-volatile-treatment (AVT). Concern for cracking in the U-bend section of the tubes has been alleviated by plugging the inner row tubes in the susceptible units.

Leaks due to stress corrosion cracking at the tube to tube support plate "dent" locations continue to occur. These leaks are not considered a significant safety hazard because the tube is constrained by the tube support plate at the dent location and will retain its integrity under postulated accident conditions. During normal operation, stress corrosion cracking will progress at a stable rate, close monitoring will result in leakage detection, and corrective actions taken.

Continued deterioration and plugging of tubes leads to a reduction of steam generator heat transfer capability and ultimately to a decrease in electric power generation capability. Some severely affected units are approaching the point where it is becoming economically infeasible to continue operation. The licensees involved, Virginia Electric and Power Company and Florida Power and Light Company, have taken steps for the possible replacement of the steam generators at the Surry and Turkey Point units. The proposed replacement program at Surry Units 1 and 2 have been reviewed by the NRC and the Safety Evaluation Report completed. Replacement of the Unit 2 steam generators is scheduled to begin in early February of 1979 and Unit 1 replacement is scheduled for the fall of 1979. Replacement programs for Turkey Point Units 3 and 4 are under review.

Westinghouse is also developing a technique for returing steam generators rather than replacing the entire component. The procedure has been performed at a prototype facility constructed by Westinghouse. A topical report regarding the procedure is expected to be submitted for NRC review in early 1979.

On September 20, 1978, Point Beach Unit 1 was shutdown when a primary to secondary leak exceeded the unit's technical specification leakage rate limit. Eddy current testing (ECT) revealed that the source of the leak was two tubes with cracks located within the thickness of the tube sheet. In addition to the two leaking tubes, ECT revealed four tubes with similar cracking within the tube sheet. This phenomenon is known as "deep crevice cracking." Early generation steam generators in which the tubes were not full-depth expanded in the tube sheet may be susceptible to this phenomenon. Because of the constraint provided by the tube sheet, the deep crevice cracks are not considered a significant safety concern during normal operation or postulated accident conditions.

Combustion Engineering

"Denting" has been observed at four CE plants: Maine Yankee, Millstone Unit 2, Palisades, and St. Lucie Unit 1. With the exception of Palisades, denting is limited to the upper "drilled design" support plates (similar to Westinghouse) in these units. The lower support plates in the Palisades steam generators are of the "drilled design" and have suffered denting. Millstone Unit 2, Maine Yankee, and Arkansas Nuclear One Unit 2, all of similar CE design, have removed lugs and portions of the solid rim in the uppermost support plates to reduce the susceptibility of the plates to denting-related cracks and tube distortion.

Denting was discovered in the upper tube support plates at St. Lucie Unit 1 during inspections conducted in April 1978. Steam generator inspections conducted in November 1978 indicated that the level of denting had increased slightly; although most of the support plate annuli were closed with corrosion products, the support plates appeared to be in good condition. Florida Power and Light Company has proposed a chemical cleaning process intended to remove the corrosion products from the tube/tube support plate crevices before the magnitude of denting becomes excessive. This program is being carefully reviewed by the NRC. Chemical cleaning is planned for April 1979.

Babcock & Wilcox

Leaks in B&W steam generators have been limited to the Oconee Nuclear Plant where the first tube leak occurred in July 1976. To date, 14 tube leaks, all at the Oconee units, have occurred in B&W steam generators. The majority of these leaking tubes were located adjacent to the open inspection lane. Laboratory examination of removed defective tubes indicated that the tube failures were caused by the propagation of circumferential fatigue cracks by flow-induced vibration The initiation mechanism for the cracks is unknown.

B&W and Duke Power Co. are investigating the possibility of eliminating this phenomenon through tube sleeving. In a demonstration program reviewed and approved by the NRC, Duke Power Company has installed a limited number of tube sleeves in the Oconee Unit 1 steam generators. The sleeves do not function as a primary or secondary pressure boundary but only as a stiffening device to reduce dynamic stresses.

On October 19, 1978, Duke Power Company informed the NRC that two steam generator tube plugs had been lost at the Oconee Unit 1 Nuclear Power Plant and were believed to be loose in the primary coolant system. The two plugs were lost during tube plugging operations in the Unit 1, B steam generator. The lost plugs are approximately 2 inches in length, 1/2-inch in diameter and 1/2-pound in weight. Efforts to locate and retrieve the plugs were unsuccessful. The NRC reviewed the safety significance of the loose plugs and determined that operation with the plugs loose in the primary coolant system was acceptable. Duke Power Company has modified their tube plugging quality assurance program and the NRC Office of Inspection and Enforcement is reviewing the revised program to ensure that the possibility of losing more plugs in the future is minimized.

NRC Actions

The NRC staff continues to closely monitor, review and evaluate, and approve the acceptability of continued operation of plants experiencing steam generator tube problems. A number of generic reviews and studies have been undertaken as part of three generic tasks in the NRC Program for the Resolution of Generic Issues. Specifically the generic Task Action Plans A-3, A-4, and A-5 are directed at the particular problems of Westinghouse, Combustion Engineering and Babcock and Wilcox.

Under these tasks generic studies will be conducted to (1) evaluate inservice inspection results from operating reactors, (2) evaluate the consequences of tube failures under postulated accident conditions, (3) evaluate tube structural integrity, (4) establish tube plugging criteria based on new information, (5) define the requirements for monitoring secondary coolant chemistry, (6) evaluate inservice inspection methods, and (7) review design improvements proposed for new plants.

On September 7 and 8, 1978, the NRC, Division of Operating Reactors, sponsored a steam generator workshop in Bethesda, Maryland. The workshop included presentations by representatives from Westinghouse, Combustion Engineering (CE), Babcock and Wilcox (B&W), and the National Laboratories involved in the Task Action Plans and a panel discussion of significant issues affecting steam generator integrity was conducted. Approximately 200 attended. The workshop provided a forum for the exchange of information throughout the industry and with the NRC.

Future reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090-6, "Report to Congress on Abnormal Occurrences: October-December 1976," and updated in a subsequent report in this series, i.e., NUREG-0090-7. It is further updated as follows:

76-16 Feedwater Nozzle Cracking in Boiling Water Reactors

Beginning in 1974, inspections at 21 of the 23 applicable boiling water reactor (BWR) plants licensed for operation in the U.S. have disclosed some degree of cracking in the feedwater nozzles of the reactor vessel at all but three of the 21 plants inspected. The exceptions were a plant with less than two years of operation at the time of inspection, a plant with welded nozzle thermal sleeves, and a plant which originally had tight interference-fit sleeves and whose nozzles were inspected and found crack free by ultrasonic means, but which has nonetheless undergone clad removal and received new sleeves. Two other facilities have not yet accumulated significant operating time and have not yet been inspected, although all will eventually be inspected. Those plants inspected to date which have exhibited feedwater nozzle cracking are as follows:

Plant Name	Licensee	Location
Browns Ferry Unit 1	Tennessee Valley Authority	Limestone County, AL
Browns Ferry Unit 2	Tennessee Valley Authority	Limestone County, AL
Brunswick Unit 2	Carolina Power & Light	Brunswick County, NC
Cooper Station	Nebraska Public Power District	Nemaha County, NE
Dresden Unit 2	Commonwealth Edison Co.	Grundy County, IL
Dresden Unit 3	Commonwealth Edison Co.	Grundy County, IL
Hatch Unit 1	Georgia Power Co.	Appling County, GA
Humboldt Bay	Pacific Gas & Electric Co.	Humboldt County, CA
Millstone Unit 1	Northeast Nuclear Energy Co.	New London County, CT
Monticello	Northern States Power Co.	Wright County, MN
Nine Mile Point Unit 1	Niagara Mohawk Power Co.	Oswego County, NY
Oyster Creek Unit 1	Jersey Central Power & Light Co.	Ocean County, NJ

Plant Name	Licensee	Location
Peach Bottom Unit 2	Philadelphia Electric Co.	York County, PA
Peach Bottom Unit 3	Philadelphia Electric Co.	York County, PA
Pilgrim Unit 1	Boston Edison Co.	Plymouth County, MA
Quad-Cities Unit 1	Commonwealth Edison Co.	Rock Island County, IL
Quad-Cities Unit 2	Commonwealth Edison Co.	Rock Island County, IL
Vermont Yankee	Vermont Yankee Nuclear Power Corp.	Windham County, VT

The feedwater nozzles, part of the pressure vessel, are an integral part of the primary pressure boundary of the reactor coolant system and form a second barrier (after the fuel cladding) to the release of radioactive fission products. All of the repaired BWR feedwater nozzles met the pressure vessel code limits, however, and no immediate action was called for. Because relatively small amounts of base metal have been removed, there has been no significant reduction in safety margins. Nevertheless, the cracking is potentially serious for these reasons:

- Excessive crack growth could lead to impairment of pressure vessel safety margins requiring more complicated repair work than simple grinding.

- The design safety margins could be reduced by excessive removal of base metal.

- The exposure to radiation of the personnel performing inspection and repair tasks can be considerable.

- The repair of these kinds of cracks can result in considerable shutdown time at the plant affected.

The reactor vendor (the General Electric Company) and the NRC have concluded from their respective studies that: (1) the crack initiation is caused by fluctuations or "cycling" of the temperature on the inside surface of the nozzles; (2) the stainless steel cladding exhibited less resistance to crack initiation than the underlying low-alloy steels, and; (3) after initiation in the stainless steel cladding, cracks can be propagated by operational startup and shutdown cycles or other operationally-induced transients. The vendor has performed extensive analysis and testing to confirm the suspected cause of the cracking and to uncover possible long-term solutions - a newly designed sleeve, removal of the stainless steel cladding, reduction of the temperature differential at the nozzle, or some combination of these. The licensees involved have increased the number and extent of inspections of feedwater nozzles with repair and reinspection where cracks were found. The vendor has advised these licensees to monitor startup and shutdown procedures in an effort to substantially reduce the time during which cold feedwater is being injected into the hot pressure vessel.

Control Rod Drive Hydraulic Return Line Cracking

In a closely related area, the NRC was informed in March 1977 by the General Electric Company that a crack had been found in the nozzle of the "control rod drive (CRD) return line" in a reactor vessel in a foreign country. The CRD return line nozzle is the opening in a BWR pressure vessel through which the high pressure water in excess of that needed to operate and cool the CRDs is returned to the pressure vessel. Later in March, the Philadelphia Electric Company reported that similar cracking had been found in the CRD return line nozzle at its Peach Bottom Atomic Power Station, Unit 3. The cracks resembled those found in the feedwater nozzles and seemed to be the result of the same kind of cyclic thermal stresses that were causing feedwater nozzle cracks. Both the foreign reactor and the Peach Bottom Unit 3 reactor are representative of a small number of BWRs which do not have a thermal sleeve in the CRD return line nozzle.

The licensee removed the cracks in the Peach Bottom CRD nozzle by grinding out the cracked area, the maximum crack depth being 7/8-inch, and returned the unit to operation with the CRD return line "valved out" and with the flow and pressure in the CRD hydraulic system modified.

Inspection of other CRD return line nozzles which incorporated thermal sleeves indicated that these sleeves may not be effective in preventing this cracking phenomenon. For example, the Georgia Power Company found a crack in the CRD return line nozzle at its Hatch Plant Unit 1, which did have a thermal sleeve. (The crack was removed, the nozzle capped, and the return line rerouted to the reactor water cleanup system.)

Actions Taken to Prevent Recurrence

A. Feedwater Nozzle Cracking

Licensee/Vendor The reactor vendor is presently completing an extensive study, involving engineering analyses and scale model tests, which has confirmed the cause of cracking and has determined well-founded and rational solutions to the problem. These solutions include but are not limited to removal of feedwater nozzle stainless steel cladding, installation of a new design interference-fit thermal sleeve/sparger combination which utilizes piston rings and concentric sleeves to protect the nozzle, and feedwater system modifications to minimize feedwater temperature fluctuations.

The licensees of operating reactors have increased the number and extent of feedwater nozzle inspections. Several have removed the cladding from the nozzles and have installed either the vendor's latest design sparger and thermal sleeve or a similar configuration designed by another vendor. Others have utilized tight-fit interference-fit designs while planning to accomplish final modifications in the future.

BWRs undergoing operating license review have generally been modified by the installation of the latest thermal sleeve/sparger combination and removal of the cladding, although some are clad with thermal sleeves welded to the nozzle safe end to assure no feedwater bypass leakage.

NRC - While awaiting the final report from the vendor, the NRC is continuing to require, as deemed necessary, inspection and local removal of all cracks during refueling outages. Pertinent licensee submittals prior to and subsequent to each refueling outage are reviewed to insure that NRC criteria are being met.

The NRC anticipates approval of the GE thermal sleeve and sparger modification as one of a number of effective designs which will serve to reduce the probability of crack initiation, and the NRC will follow the formulation and implementation of hardware and procedural modifications which would serve to substantially reduce the time during which the vessel is exposed to low-temperature, low-flow feedwater. Upon submittal of the vendor's final report documenting completion of the engineering studies, the NRC will publish final guidance to licensees and applicants.

B. Control Rod Drive Hydraulic Return Line Nozzle Cracking

Licensee/Vendor - Although the original vendor recommendations involved either "valving out" or rerouting the control line such that the continuous flow of cold water to the vessel nozzle would cease, the latest recommendation is to simply remove the return line and cut and cap the nozzle at the vessel exterior. The related changes in system operation have prompted the vendor to perform control rod drive system component tests to assure continued operability under adverse conditions. Also, because of NRC questions about the amount of water which can be directed to the vessel through the control rod drive seals as opposed to that which could be obtained with the return line intact, the vendor has stated that he intends to perform analysis as soon as possible to verify that substantially the same flow is available. Meanwhile, several licensees are operating "valved out," others have re-routed, some have retained the original configuration after finding no cracking during dye-penetrant inspections, and one (Cooper) has cut and capped the return line, without re-route, in accordance with the latest vendor recommendation.

NRC - The NRC will follow the vendor testing, especially with regard to vessel return flow capability and the long-term operability of rod drive system components with the return line removed. In the interim, further licensee requests to remove the return line will be denied. However, license applicants for later BWRs will be granted such permission but must verify return flow capability and system operability by confirmatory testing prior to final approval being granted.

Future reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090-10, "Report to Congress on Abnormal Occurrences: October-December 1977," and is updated as follows:

77-8 Generic Design Deficiency

In August 1977, the NRC was informed that five facilities had a potential deficiency in the design of the Containment Recirculation Spray (CRS) system and Low Head Safety Injection (LHSI) system pumps. These facilities are the North Anna Units 1 and 2, Surry Units 1 and 2 (both operated by the Virginia Electric Power Company-VEPCO) and Beaver Valley Unit 1 (operated by Duquesne Light Company-DLC). It was determined that the net positive suction head (NPSH), calculated to be available to the pumps of these systems, is insufficient with respect to the required NPSH specified by the pump manufacturer for the intended pump operation.

The net positive suction head is a way of defining the pressure at the inlet of a pump. If this pressure is too low, cavitation can occur (i.e., some of the water will vaporize or turn to steam) and the pump may not operate correctly. Potential exists for pump flow to be low and for mechanical pump damage. The inlet pressure is determined by the pressure of the fluid reservoir from which the pump gets its water, and by the flow through the intervening piping. The acceptable pressure, and therefore the acceptable NPSH, is determined by the pressure, flow and temperature of the water and by the pump characteristics.

Both the CRS and LHSI systems are engineered safety features whose functions are the mitigation of consequences of a postulated loss-of-coolant accident (LOCA), a low probability event. The CRS system is designed to remove heat from the containment in order to reduce the containment pressure to below atmospheric pressure within one hour after a postulated LOCA. It consists of four subsystems, each with 50 percent capacity. The pumps take suction from the containment sump, with two pumps located inside and two pumps located outside the containment. The LHSI system is designed to inject cold borated water into the reactor core. The system consists of two 100 percent redundant and independent subsystems. Initially the system is connected to the Refueling Water Storage Tank (RWST), but is switched to the containment sump when the RWST reaches a low-low level.

For each of these systems to satisfy its intended safety function, the pumps in each system must be capable of providing the design flow rate under all postulated post-LOCA conditions of containment pressure and pump water temperature. Thus, conditions leading to inadequate NPSH for the CRS and LHSI pumps for extended periods of time could affect the capability of these systems to perform their intended safety function.

Actions Taken to Correct Deficiencies

The North Anna Units 1 and 2 were in the first stages of receiving an operating license review when the deficiency was discovered. The solutions proposed by VEPCO have been approved by the NRC as final design changes and operations.

For the CRS system:

1. To assure adequate NPSH to the inside CRS pump, 150 gallons per minute (gpm) of quench spray (QS) system water will be diverted to the suction side of each pump. The cold QS water will lower the vapor pressure of the water entering the pump.

2. The outside CRS pumps will be operated at the full 3640 gpm flow rate but will require addition of cold water at the pump suction from a new casing cooling subsystem. The new system is capable of injecting 800 gpm of 50°F water into each outside CRS pump suction.

For the LHSI system, VEPCO has shown by analysis and test results that the LHSI pumps require no modification and by cross connecting the outside CRS and the LHSI system, emergency core cooling backup is provided.

All of the North Anna Units 1 and 2 final systems modifications have been approved. Additional information is contained in Supplements 8 and 9 to the Safety Evaluation Report for an operating license and in Amendment 5 to the Unit 1 license.

Surry Units 1 and 2 and Beaver Valley Unit 1 have installed NRC approved interim design changes to minimize NPSH deficiencies until a final solution is approved and can be installed. The design and operating changes in effect at Surry Units 1 and 2 include the following:

1. Installation of flow-limiting orifices in the discharge lines of the two CRS system pumps located outside containment. These orifices reduce the flow from 3300 gpm to 2250 gpm and reduce the required NPSH to 6.4 feet compared with a calculated available NPSH of 7.3 feet.

2. The CRS system pumps located inside containment will operate in a cavitation mode only for a limited time (from 11 to 35 minutes after a postulated LOCA) and at a reduced flow rate of 3000 gpm.

3. Limits have been placed on certain operating parameters (service water temperature, containment temperature, minimum refueling water storage tanks volume and containment air partial maximum pressure) to ensure the validity of the assumptions made in the calculation of the available NPSH.

4. With respect to the LHSI system, a potential for pump cavitation was found to exist for a short period of time during the recirculation mode if the flow rate exceeds 3500 gpm. In order to assure that this flow rate, which is adequate for long-term core cooling requirements, will not be exceeded, VEPCO will throttle the valves in the pump discharge line while monitoring the flow rate in the control room to ensure that the flow rate is limited to 3500 gpm.

The design and operation changes in effect at the Beaver Valley Unit 1 include the following:

1. To assure an adequate amount of NPSH for the CRS pumps outside containment, 250 gallons per minute (gpm) of cold quench spray (QS) water from each QS header will be diverted to the sump area at that point where water is drawn to the outside CRS pump suctions. The cold QS water will lower the vapor pressure of the water entering the pump.

2. The CRS pumps located inside containment will operate for about 13 minutes in a mild cavitating mode with a reduced flow rate of 3000 gpm. Pump test results have demonstrated that this mode of operation will not damage the pump.

3. The discharge valves will be partially closed during the recirculation phase to reduce the total flow rate from 4200 gpm to 3100 gpm. The reduced flow rate does not cause LHSI flow to be less than the minimum required for emergency core cooling in either the short term or the long term.

4. An additional 17,000 gallons capacity has been added to the refueling water storage tank (RWST) to provide further assurance that adequate NPSH is available to support 3100 gpm flow without cavitation.

Final design changes and modifications have been proposed for the Surry Units 1 and 2 and the Beaver Valley Unit 1. The proposed final modifications include combinations of diverting cold water to the suctions of inside and outside CRS pumps, limiting flows by using orifices and cavitating venturis, and adjusting RWST and other inventories. The NRC staff is nearing completion of these reviews.

Future reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0900, Vol. 1, No. 2, "Report to Congress on Abnormal Occurrences: April-June 1978," and is updated as follows:

78-2 Fuel Assembly Control Rod Guide Tube Integrity (A Generic Concern)

As reported previously, examination of fuel assembly control rod guide tubes after service in several operating pressurized water reactors (PWRs) disclosed significant amounts of wear. At the extreme, some tubes had been worn through showing sizeable holes. The cause was found to be flow-induced vibration of fully withdrawn control rods. The rod tips, vibrating against the guide tubes, induced degrading wear, probably aided by some corrosion mechanism.

The safety significance of the incidents relates to the functions of the guide tubes. They serve both as fuel assembly structural members and as channels for control rod movement. Thus, guide tube failure could adversely affect either the maintenance of a coolable core geometry or the scram capability of the control rods, or both.

Evaluation of guide tubes assuming the maximum amount of wear observed established that structural integrity was maintained. Both analytical and experimental results showed that normal and accident loadings could be sustained.

Although the observed severe wear thus far has been confined to facilities designed by Combustion Engineering (CE), the potential for such wear in Westinghouse and Babcock and Wilcox plants and in Exxon Nuclear fuel assemblies is under investigation by the NRC staff.

Update to Previously Reported Corrective Actions

Licensee/Vendor - Extensive inspections were conducted at all CE plants. Discharged fuel assemblies kept in spent fuel pools were examined promptly. Assemblies in operating reactors were examined during regularly scheduled refueling shutdowns.

Concurrently, a program of testing and analysis was conducted by CE. Tests showed that flow-induced vibration of control rods was the principal factor. To overcome the susceptibility to wear by the guide tube material (Zircaloy-4) and to recover the design margin lost by wear, stainless steel sleeves were designed and installed in all worn guide tubes and in any tube, worn or not, scheduled for locating under a control rod after refueling.

Prior to installation of stainless steel sleeves during a refueling outage, operators of CE reactors instituted the practice of inserting the control rods three inches further into the core than the normal fully withdrawn position. That action both reduced the local wear intensity and provided added assurance of scram capability.

NRC - The NRC staff has maintained close liaison with representatives of the licensees and vendors to discuss problems related to this issue. All proposed programs have been reviewed prior to taking action at any facility to assure continued safe operation. Approval was granted both to operate with the control rods inserted three inches further into the core and to install, and operate with, stainless steel sleeves. The staff has required that all inspection programs be submitted for review well in advance of refueling shutdowns.

Additional Corrective Actions

Licensee/Vendor - Further inspection results have shown that what had been identified earlier as the worst amount of wear was not exceeded. Thus, the analyses showing worn tubes to be structurally adequate remains acceptable. Also, tubes associated with rods repositioned three inches lower than the fully withdrawn position generally exhibited less wear at the lower location, showing that the deep insertion plan had been effective.

Inspection of fuel assemblies during the refueling shutdown at Baltimore Gas and Electric Company's Calvert Cliffs Unit 2 showed guide tube wear patterns very similar to what had been observed at the sister plant, Calvert Cliffs Unit 1. The wear severity was somewhat less, however, because the plan to insert the scram rods three inches deeper had been put in place about midway through the fuel cycle.

Additional out-of-reactor hot loop testing by CE showed the important role of flow-induced vibration of the control rods in the guide tube wear problem. The vibration and hence, the wear, was lessened by reducing some of the guide tube coolant (water) flow. Two fuel assembly modifications were designed to reduce the coolant flow. One involved inserting a splined cylinder in the top of the guide tube. The second involved reducing the size and number of flow holes in the bottom of the guide tube. Both modifications, in limited number, are installed in currently operating cores to verify the loop test results. Test results favored the modified flow hole design.

NRC - The NRC has closely followed the analyses and experiments performed by CE. The NRC staff is in substantial agreement with the vendor that the results point to control rod flow-induced vibration as the principal factor in guide tube wear. Therefore, design modifications intended to reduce flow in the guide tubes were judged appropriate. The NRC has approved the modified designs for limited operation on the basis that they will mitigate the wear problem. Final approval of either design modification as a solution to the problem will be contingent on the results of further out-of-reactor experiments and examination of the modified assemblies now in operation. The operating experience, post-operation inspection and evaluation will require about one more year.

Also, inspection programs must be completed to evaluate the performance of fuel assembly guide tube wear sleeves in support of continued reactor operation. The first opportunity to examine and evaluate sleeved guide tubes will occur during the refueling outage of the Millstone Nuclear Power Station Unit 2 in the spring of 1979. The NRC staff has begun an evaluation of the proposed inspection program for that plant. Continued operation of other CE plants with sleeved guide tubes depends, in part, on the outcome of the Millstone inspections.

Future reports will be made as appropriate.

* * * * * * * *

The following abnormal occurrence was originally reported in NUREG-0090 Vol. 1, No. 3, "Report to Congress on Abnormal Occurrences: July-September 1978," and is updated as follows:

78-4 <u>Degraded Primary Coolant Boundary in a Boiling Water Reactor</u>

As previously reported, the licensee (Iowa Electric Light and Power Company) discovered a leaking through-wall crack in a nickel alloy (Inconel) fitting, called a "safe end" in the Duane Arnold Power Plant located in Linn County, Iowa. The safe end is a short transition piece (approximately 8 inches long) joining a section of primary coolant recirculation line piping to the reactor vessel. Nondestructive testing of the other seven identical safe ends revealed that all had indications of cracks or weld irregularities; however, these flaws did not penetrate to the surface of the safe end.

The licensee has removed all eight safe ends and replaced them with safe ends of an improved design. The new design minimizes the tight crevice formed by the fit up of the safe end and an internal thermal sleeve; such crevices are known to enhance the possibility of stress corrosion cracking in an adverse chemical environment (e.g., in stagnant oxygenated water).

The safe end installation was completed in December 1978, and the licensee initiated a testing program leading to resumption of plant operation. On January 8, 1979, the NRC issued an amendment to the Duane Arnold license approving the design of modified safe-ends and authorizing the unit to return to power following completion of certain audit and test requirements. The audits included ultrasonic and radiographic examination of all repair welds. During the audit, surface irregularities were identified in the radiographs of the root passes on the pressure boundary welds. Several meetings, discussions and on-site inspections were held with the licensee and his contractors to evaluate the effect of the surface irregularities in the weld root inserts. The evaluations included extensive stress and fatigue analyses and corrosion behavior analyses assuming "worst case" conditions. The evaluations concluded that the welds were acceptable for the potential service conditions.

(Editor's note: During the final editing of this report, the following additional information became available.)

While preparing for a cold hydrostatic test prior to operation, the licensee determined on January 28, 1979 that there was flow blockage in either the N2B riser or in the associated jet pumps numbers 3 and 4. Fiberscope inspection revealed that a lead shielding plug had not been removed from the N2B nozzle prior to closure of the inlet piping. This plug consisted of a thin aluminum and carbon steel can filled with shaped lead blocks. During the preparations for the leak test, water flow in the line pushed the plug into the jet pump assembly where it came apart. Retrieval operations recovered all 10 of the lead blocks and most of the can. A small fragment of the .016" thick aluminum backing plate and 16 small .015" thick carbon steel tabs from the can were not recovered and were assumed to be in the reactor vessel.

On March 5, 1979, the NRC issued an amendment authorizing the Duane Arnold facility to return to power in three incremental steps (to remove any possible lead contamination and to dissolve the piece of aluminum). The amendment also changed the Technical Specifications to incorporate augmented inservice inspection of the repaired safe-ends.

The Duane Arnold facility achieved criticality on March 6, 1979 after an outage of almost 9 months. The hot (500°F) hydrostatic test was successfully completed on March 7, 1979. The licensee expects to resume normal power operations.

The NRC has concluded that the crack in the recirculation inlet safe end at Duane Arnold was likely due to a plant-unique condition and that it does not have generic implications beyond the few safe end designs of similar geometry. The particular cracking problem at the Duane Arnold Power Plant is also a part of the overall concern for cracks in pipes at boiling water reactors, as discussed in Abnormal Occurrence No. 75-5 of NUREG-75/090, "Report to the Congress on Abnormal Occurrences: January-June 1975," and updated in subsequent reports in the series, i.e., NUREG-0090-1, 2, 3, 5, and Vol. 1, No. 3. Therefore, in order to avoid duplicate reporting, any further developments pertaining to the Duane Arnold Power Plant cracking problem will be reported through updates to Abnormal Occurrence No. 75-5, with Abnormal Occurrence No. 78-4 considered closed for purposes of this report.

APPENDIX C

OTHER EVENTS OF INTEREST

The following events are described below because they may possibly be perceived by the public to be of public health significance. Neither event involved a major reduction in the level of protection provided for public health or safety; therefore, they are not reportable as abnormal occurrences.

1. Broken Seals on Four Containers of Highly Enriched Uranium Exported to Romania

   The General Atomic Company, San Diego, California, has been authorized by the U.S. Nuclear Regulatory Commission under Export License XSNM-885 to export 38.92 kilograms of highly enriched uranium (HEU) and 43.47 kilograms of low enriched uranium as TRIGA reactor fuel from the United States to Romania. The licensee is exporting the HEU as multiple shipments, each containing less than 5 kilograms. Shipments of less than 5 kilograms are exempt from the controls prescribed by 10 CFR 73.30. For HEU export shipments under this license, U.S. Government seals are to be affixed to each container of HEU.

   The initial shipment (contained in four drums) consisted of 100 fuel elements containing a total of 4.5 kilograms of highly enriched uranium. A unique serial number is imprinted in each element.

   The packaging and sealing of the four containers which constituted the initial shipment were observed by an NRC inspector at the licensee's site. (Three-strand seal wire was used rather than the minimum 19-strand recommended by a Regulatory Guide). When the seals were subsequently examined on December 16, 1978 at J. F. Kennedy Airport in New York prior to export, all seals were found to be broken. Based on examination of the four containers at Kennedy Airport, the NRC inspectors decided that the contents of the containers had not been disturbed. Consequently, the NRC inspectors resealed the containers without opening the containers to verify the contents, and the containers were sent on to Romania. Subsequently, in a special inspection arranged through the Department of State, the International Atomic Energy Agency (IAEA) inspected the containers and their contents on January 5, 1979. It was noted that three of the seals affixed at Kennedy Airport were found to be intact while one of the wires was no longer intact within the fourth seal; somehow, the wire became detached inside the seal button. (An NRC inspector examined this seal, after it was returned by IAEA, and determined that the wire had broken inside the seal button.) The IAEA verified that the contents of the containers were as shipped from the General Atomic Company.

Procedures have been developed and implemented for proper inspection and verification of the remaining material for Romania prior to export.

2. Special Safeguards Review at Uranium Fuel Processing Facility

On October 30 - November 2, 1978, the NRC initiated a Special Safeguards Review to evaluate the capability of the safeguards system at United Nuclear Corporation, Wood River Junction, Rhode Island, to defend against the hypothetical insider threat. This review was conducted as a result of the inspection and investigation of allegations regarding guard qualifications and the inadvertent shipment of 68 grams of SSNM to the UNC Montville, Connecticut facility. The investigation report concerning the allegations on guard record falsification is under review for possible referral to the Department of Justice. The inspection of the inadvertent shipment of SSNM has been completed. The NRC is presently considering escalated enforcement action based on the findings of the investigation and inspection discussed above.

The report of the Special Safeguards Review Team was hand delivered to the licensee on December 11, 1978, along with a letter directing the licensee to respond to the concerns expressed in the report within twenty (20) days. The licensee responded on January 3, 1979.

The NRC did not regard any of the items identified in the Special Report as requiring emergency action. The term "emergency action" is used by the NRC staff to indicate actions required when deficiencies exist that would make a facility so vulnerable to the hypothetical threat that continued operation of the facility would be inimical to the common defense and security or would pose an undue risk to the public health and safety.

The licensee stated in the January 3, 1979 letter that corrective action was completed on five of the six items described in the report as having "high importance" prior to November 30, 1978. Subsequently, corrective action was taken on the sixth item.

The NRC has indicated that when the corrective actions contained in the licensee's letter of January 3, 1979 have been accomplished satisfactorily, the licensee can again be considered to have a high assurance against the hypothetical threat.

The NRC performed an inspection the week of January 22, 1979, and has confirmed that the licensee has completed corrective action on all items which did not require equipment procurement and installation.

l six items described as having "high importance" were confirmed to be completed. The NRC has requested additional corrective action on one item. The licensee's action on this item will be inspected by the NRC.

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

MEMORANDUM FOR: B. C. Rusche, Director
Office of Nuclear Reactor Regulation

Ernst Volgenau, Director
Office of Inspection and Enforcement

R. B. Minogue, Director
Office of Standards Development

K. R. Chapman, Director
Office of Nuclear Material Safety and Safeguards

SUBJECT: DEVELOPMENT OF PERFORMANCE EVALUATION PROGRAMS

The subject of collection, analysis and uses of performance data from components and systems in operating reactors (and other nuclear facilities) has been discussed within NRC for several years. The Licensee Event Reports (LER) system is well established and the Nuclear Plant Reliability Data System (NPRDS) now has 35 reactors reporting some failure data for safety related components and systems. What is generally lacking is a systematic process for analyzing these data into a form which can be utilized in the regulatory process. Such feedback has several potential benefits, e.g. place our decision-making on a more factual basis, provide motivation to our licensees to contribute information to NPRDS. Each of your Offices has a need for different feedback from operating reactors, e.g. NRR needs to know the field performance of diesel-generators, IE need guidance on which systems require more or less inspection.

In order to meet such needs, MIPC and RES have agreed that:

1. MIPC will take responsibility for the collection of basic data, e.g. NPRDS and LERs, and the analysis of such data.

2. RES will take the responsibility for the development of reliability models for MIPC's use in such analyses.

In order to be responsive to your needs, we need to establish what kind of data should be collected and to what form it should be analyzed. Different uses of the data require different analyses. For example, risk assessment and preparation of Technical Specifications may require outage time for maintenance in addition to the failure rate for a specific component. However, for inspection requirements, the trend of failure rate may be more important. In order for operating data to contribute to decision making, one also needs to establish decision criteria for such data, e.g. should operating restrictions be considered when diesel generator availability falls below 99%, 90% or 80%? The criteria also need to be considered within the context of the overall plant risk. The forms of such criteria influence the types of data which are collected and the reliability models which are used in the analysis of the data.

We propose that a Task Force be established to:

1. determine what analysed data is needed by each Office,

2. determine what reliability models are needed or the programmatic work required for the determination of such models, and

3. recommend either potential decision criteria or what programmatic work should be performed to establish decision criteria.

We recommend that the Task Force be required to submit its report by June 30, 1977. With your concurrence, we propose that Dr. William Vesely of RES and Mr. Richard Hartfield of MIPC be appointed co-chairmen of this Task Force.

We invite your comments and designation of your representative on the Task Force.

Saul Levine, Acting Director
Office of Nuclear Regulatory Research

W. G. McDonald, Director
Office of Management Information and
Program Control

cc: S. H. Hanauer

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Exh. #6

NOV 6 1974

00050

Westinghouse Electric Corporation Power Systems

PWR Systems Division

Box 355
Pittsburgh Pennsylvania 15230

November 1, 1974

NS-RS-420

Mr. Saul Levine
Project Staff Director
Reactor Safety Study
U.S. Atomic Energy Commission
1717 H Street, N.W.
Washington, D. C. 20545

Dear Mr. Levine:

As you requested in your letter transmitting a draft copy of WASH-1400, we have reviewed the report and attached our comments.

The report represents an important contribution to the understanding of the risks associated with nuclear power. It helps demonstrate that the nuclear industry has achieved a significant level of safety in the design of nuclear power plants. The thoroughness of the Reactor Safety Study efforts in assembling and integrating the information needed to evaluate these risks is to be commended.

It is our opinion that the report can be useful in licensing activities of a generic nature. It would be unfortunate if the report were not used to assist industry and regulatory bodies in establishing criteria. We concur with your statement that the techniques utilized by the Reactor Safety Study group need further development before they are used in individual, rather than generic, applications. We believe that these techniques are inappropriate for individual application evaluations at this time.

If you have questions on any of our comments, please do not hesitate to contact us.

Very truly yours,

Romano Salvatori

R. Salvatori, Manager
Nuclear Safety Department

RS:jmb

Attachment

POOR ORIGINAL

WASH 1400 COMMENTS
SUMMARY REPORT

1. Figures 1, 2 and 3 are good, applicable representations of the conclusions of the report. They should be highlighted in the Appendices and utilized in public discussions.

2. Table 1 is inconsistent with other similar tables in the report. (e.g. see p. 19.)

3. The Q&A approach is almost self-defeating. Answers are too wordy and often misleading or confusing. For example, 2.1 discusses staffing. The figures are hard to reconcile. Q/A 2.2 wanders from the stated subject to discuss the "fundamentals" of nuclear power conversion. Q/A 2.5 - the first sentence is inappropriate. Only when the release is made are the radioactive products a problem to society. Properly contained, they pose <u>no</u> risk. Again, the answer wanders. Q/A 2.7 - the response never directly addresses the question. This question, in particular, needs a direct firm response. Q/A 2.8 - wanders before addressing the question posed. Q/A 2.9 and Q/A 2.10 are revisions of 2.7. Is this repetition intentional?

4. In Q/A 2.11, a probability of 1:17,000 is given as the <u>total</u> likelihood of a core melt. (Considering contributions from all accident sequences.) Elsewhere in the report, 1/17,000 is stated as the probability of the <u>most likely</u> core melt sequence. (e.g. see Q/A 2.13, Q/A 2.16, Q/A 2.17, Q/A 2.18.) This inconsistency is serious and must be corrected.

5. Q/A 2.12 - the discussion of health effects and latent problems should be improved by rewriting to more directly address the subject.

POOR ORIGINAL

WASH 1400

A number of factors are stated (page 239) that make the model conservative, but all are associated with consequence end, and none address the likelihood or accident probability end. Some related to accident likelihood that are conservative are:

1. Generally equipment performance less than that to meet the minimum SAR requirement constituted failure and was assumed to result in core melt.

2. Some transient initiated sequences were assumed to result in core melt although further analysis could show otherwise.

3. In general little or no credit was given to possible operator backup corrective actions during the course of an accident.

4. Operator error rates in following emergency procedures were selected to not be optimistic, i.e., high stress assumed.

5. Fairly large uncertainty bands about best estimate failure rates were incorporated and carried through.

6. The log normal approach of combining failure probabilities used the larger of the 90% error factors where 90% bounds were not symmetric.

7. The smoothing procedure allocates a probability from each release category to the higher release categories.

These should be clearly identified as factors which make the results conservative rather than realistic.

On page 127 the PFHSVHR is applied as the initiating event and should be based on the annual probability.

As presently calculated, it appears to be based on a period of one month for valve failures and one year for pipe failures. In conjunction with this, we feel that 1×10^{-8}/hr is a factor of 10 too high for serious rupture of a single valve (see Appendix III comment).

2. A feedline rupture in vicinity of AFW connection has potential to spill water from all 3 AFW pumps until operator isolation of spillage is complete.

3. Page 120. SOV-102 is assumed to be out 19 hours every 4.5 months; this seems high and is not consistent with the App. III 7-hr. valve assumption.

4. Page 124 top of page. There appears to be no credit given for the possibility of replenishing AFW supply from the 300,000 gal. tank. In view of the time available we feel the chance (10^{-3}) of not getting the fire main valves open is too high. The AFW system is frequently used, therefore, the normal makeup path must have valves that are occasionally opened. The turbine pump should not be assumed to be unavailable after 8 hours.

Containment Spray

1. Page 161. The subtraction estimate to account for CMF double accounting should really subtract 2.6×10^{-4} instead of 1×10^{-4}, i.e.,

$$\left(\sum p_i\right)^2 - \left(\sum (p_i - p)\right)^2 = 2p\sum p_i - p^2$$

$$\neq \left(\sum p_i\right)^2 - p^2$$

COMMENTS ON TABLE 5.2 OF MAIN REPORT

1. The sequence $S_1D-\epsilon$ should be 3×10^{-6} to be consistent with the 9.5×10^{-3} ECI unavailability of Table II-3.

2. Some explanation is needed for the values used for δ. We understood it to represent containment leakage which has 2×10^{-4} or 2×10^{-3} median values from Table II-3. For the transients 2×10^{-4} was used, but for LOCA's the value appears to be more like 5 or 10×10^{-3}.

3. The other containment failure mode probabilities don't always appear consistent with values from Table 2 - Appendix V. AF - α, AG - α, AHF - δ, ACHF - ϵ. We assume tha the same containment failure mode probabilities determined for A s ces are used in S_1 and S_2 sequences but this is not stated. This also seemed to be the case for the transient sequences except for TMLB' - δ.

4. This is an important table and we feel the interested reader should be able to reproduce all the probabilities from appendix information. In this regard there should be consistency between the various tables. It would also be helpful if a table were included in one of the appendices summarizing those cases where the probabilities could not be directly multiplied because of dependencies. For example in the AD sequences we assumed that D was effectively doubled because of the estimated 0.01 RCP flywheel contribution. On the other hand for $S_1D-\epsilon$ we assume there is a factor of 2 error. Similarly, for the multiple system sequences (CD, HF, CHF, DF, DG and possibly the β cases) the combined result and principle dependencies could be listed.

5. Since loss of all electric power has been included separately as a LOCA sequence (e.g., AB), we assume this fault is not included in sequences like AHF.

APPENDIX II, VOLUME I

1. The auxiliary feedwater 0 - 8 hr. without net Q_{upper}, Q_{lower} and Q_{median} are a factor of two greater in Table II-1 than calculated on page 109 of App. II, Vol. II.

2. The high pressure injection point estimate does not lie within the range of SAMPLE results given. Our SAMPLE analysis based on material presented in App. II, Vol. II gives results about half as large, hence we suspect a change has been made which is not reflected in Table II-1.

3. The low pressure injection result similarly seems higher than would be expected from point estimates given.

4. The auxiliary feedwater system results (1.5×10^{-4}) for the transient (no LOCA) should be included in Table II-1.

5. In some cases the hardware contribution includes human error, this should be noted in Table II-1.

APPENDIX II, VOL. 1

Page 35 Section 2.2.3, line 5 - "Figure II-4" should be "Figure II-5"

line 9 - "CLCS" should be "CLS" to be consistent with the fault tree.

Page 36 II-6' line 4 - "Figure II-4" should be "Figure II-6"

line 6-7 - the meaning of the phrase "... and two input transfers for the only input "Failures In Pump Drive Cause Pump to Fail to Start" is not clear.

Page 45 line 3 - there seems to be a word missing from the phrase "... single or active events)...".

Page 60 line 17 - add a period after "...equation..."

Page 23 The definition of a primary component fault state is acceptable as is, but definitions offered for secondary and command deserve a comment.

A secondary failure is more associated with the environment or loads that a component is exposed to. If a component is subjected to conditions beyond which it was designed for, it should be expected to fail. This means that a secondary failure is really the probability that the component is subjected to conditions beyond its design limit. The command failure is unnecessarily confusing since it does not apply directly to the component. It is what is commonly referred to as a secondary event. Such distinction between failures is not really necessary.

Page 36 The event naming described on this page could be useful most particularly when compiling a data bank. However, transfers should be used to identify identical components within a fault tree.

It should not be left to the reader to note that a particular event name appears in more than one place in a fault tree.

Page 43-44 Item 3 Does this mean that only second order failures need to be postulated?

Item 5 Human interfaces can occur at many places. Are human interfaces limited to required action or are all possible interfaces postulated?

Page 45 Item 3 Care must be taken what when several events are combined and represented by a single event, all the events must be independent. In addition when n events are combined, as example on piping, is the failure rate that of a single event or n x failure rate of a single event?

Page 55 Partial failures can not be postulated usefully unless partial failure rates are available and the affects of partial failures on the system are considered.

Page 57 By summing the system failure probability and the system unavailability to obtain the total system probability, common failures are not accounted for correctly.

In general the subject Appendix should prove to be a valuable handbook that wi' aid in standardizing fault tree analyses.

APPENDIX II (VOL. 2)

RPS

1. Page 93 - The assumptions made regarding trip breaker test and maintenance outages amount to 108 hours per year when one or the other of the breakers is not in service. We assume that you have checked records indicating this is a reasonable (T&M) unavailability during power operation.

2. Shouldn't the terminal board short (ITM008Q - 25Q) identified on pages 92, 96 and 100 have a failure rate of 1×10^{-8}/hr. instead of 3×10^{-7}/hr (see last line of page 105 Appendix III)? If so, the test and maintenance contribution would reduce to 4×10^{-7} and the doubles hardware contribution to 1.85×10^{-6}.

3. Shouldn't the bypass breaker fault (ICB0004C & 5C) be either combined with a human error to fail to correct, or be based on a shorter exposure time?

4. ATWT parameter studies have shown that there is considerable reduction in pressure peak even if only a small fraction of the rods trip, hence assumption of more than two rods sticking is conservative.

AFW

1. Page 121 and 127. There appears to be an inconsistency in logic for the fault PPPMSVHR constituting pipe and valve ruptures in the MSVR. On page 121 we believe the exposure time should be more like the first 8 hours after the accident plus a relatively short period before the accident because if it is sufficiently severe to cause the damage cited it should be detected and corrected or the pla t shutdown.

2. The CSIS would seem to be one example where operator action to backup certain faults could reasonably be expected, e.g., actuation of components, taking equipment out of test mode and placing in operation, opening blocked path.

CLCS

1. Page 190 - The treatment of NCNCA120, the never tested contact fault, looks questionable. Use of the failure rate and averaging over 40 years would give an average value for Q_H (page 193) about a factor of 15 higher than the point estimate shown.

2. The statements on pages 192 and 193 seem to indicate that if train A is in test and train B fails, CLCS still actuates. Is this correct, or does it mean train A automatically trips if train B gets signal?

Accumulator

1. Page 257 and 261. The pipe ruptures APPPH16R and APPSI46R were based on a half year detection time. This should be much shorter to overlap with a LOCA, i.e., if they were significant they would be noticed (accumulator level and pressure) and corrected or the plant shutdown.

2. We believe 10^{-5}/demand is more reasonable than 10^{-4} for check valve failure to open or close.

LPIS

1. Page 290 (top of page). This treatment implies that each of the 4 MOV's is actually out of position (closed) 0.22 times per month (once every 4.5 months). Shouldn't the 0.22 factor be replaced by a much lower expected frequency for inadvertent closure, making this contribution negligible?

HPIS

1. Page 339. The fault FCVS236, check valve fails to open, has been assumed to be 1×10^{-3} per demand on the basis that the check valve is never tested. Elsewhere, consistently the check valve failure to open has been treated as a demand value (1×10^{-4}) instead of a time dependent value. We feel the chance of a check valve failing to open or close is more like 1×10^{-5}/demand.

2. Page 359. There is a typo for fault FOL115CB which should have the same failure rate and fault exposure as FOL115EB.

3. For a small LOCA we believe that certain faults such as opening a closed valve can be corrected by operator to achieve a higher effective reliability than when considering equipment alone.

APPENDIX III

1. On pages 147-153 a quantitative human reliability analysis is given for switch to recirculation. The analysis is however incomplete and it is unclear as to how a 3×10^{-3} probability was eventually obtained for this fault. For example on page 153 the total estimated failure rate for step 4.8.1 is $10^{-2} \times 0.75 \times 10^{-1}$ -= 0.00075. However the 0.25 chance that some other (than MOV-1863 A&B) pair of switches would be selected is not carried through nor is the contribution for steps 4.9.1 and 4.9.2.

2. Pages 123-126 - The general error rates of 0.9 after 5 min., 10^{-1} after 30 min., and 10^{-2} after several hours sound unbelievably high. We do not agree that stresses and/or error rates in the post - LOCA event are similar to the inflight emergencies stated or to the motar shell example. In fact for some steps a factor of 10^{-2} was used in the recirculation switching example for times prior to 30 min.

3. Pages 96-100 and 113-118 - As the test and maintenance outages are a significant contribution to fluid system unavailability, utilities should verify that the application of these durations is consistent with procedures and maintenance experience on standby safety systems during power operation as opposed to normally operational systems. In particular maintenance of standby pumps and valves would be expected at lower frequency than if frequently in use and major maintenance would probably usually occur during plant shutdowns. Also, the method of treating test and maintenance in SAMPLE analyses should be stated, i.e., whether mean test and unavailability is used or median plus error spread.

4. Page 129 - The human error rates used for non stress situations seem very high, for example, the 10^{-2} general human error of omission would seem to indicate a large number of valve-out-of position failures should be discovered each year in safety systems. For

example, with 30 operating plants with 20 valves each being manually operated each month, on the order of 70 such reports would be expected annually or about 6 each month.

5. Pages 94 & 95 - The 1×10^{-8}/hr failure rate generally used for valve rupture seems too high for a serious rupture of a single valve. We believe that the valve rupture failure rate would be more similar to that of a pipe section, 1×10^{-9}/hr. We also believe that the chance of a check valve failing to open or close is related to cyclic usage and is about 10^{-5} per cycle. On this basis we feel the 3×10^{-7}/hr rate for a check valve failing to close is reasonably for a case in which the valve is cycled on the order of once per day but, for a check valve cycled only once per year we believe the failure rate should be much lower.

6. In Section 6.4 substantial data is developed regarding rate of pipe rupture. However, it appears to be applied to a pipe length of 17,000 feet (10% of 170,000) for the reactor coolant system pipe. This seems to be about an order of magnitude too high for the LOCA sensitive pipe length and can result in gross over prediction of the LOCA frequency. In general, the data sources referenced seem to yield consistent results with the exception of the 4 process piping failures mentioned at the bottom of page 180. Assuming these refer to the first, third, fifth and tenth items identified on page 59, note that the only rupture was actually a rubber expansion joint and not a pipe, and that the other three are identified as cracks. The severity of these cracks should be examined for significance before biasing LOCA rate bounds on this data.

APPENDIX IV

1. We were pleased to note the direct and serious-minded approach to common mode failure taken in the WASH-1400 study. This represents, to the best of our knowledge, the first real attempt to quantitatively bound the problem of common mode failure is a matter of broad proportion and that there are no simple answers which give simple solutions to the problems. The bounding estimates are the subject of our particular interest and concern.

   The techniques of failure coupling and combination bounding described in Appendix IV to the draft report are significant and useful additions to the state of the art in failure analy is and system reliability prediction. However as we understand their application they appear to represent extremely conservative estimates of the effect of common mode failure. That is to say that experience in all sorts of redundant systems indicates that common mode factors are not so severe a limitation on system reliability that they completely eliminate the benefits of redundancy as would be shown by these bounding techniques. We find it interesting that the summary results do not show a significant impact in reactor protection system reliability even when very conservative bounding techniques are applied. It is not completely clear how the conservative bounding techniques were incorporated with the Monte Carlo program sample, that is SAMPLE, but we are concerned that the conclusions and summary statements should emphasize the extreme degree of conservatism in these bounding methods. It is probably the only thing which can be done currently by way of quantitative prediction of common mode failure effect and as such we believe it is important. However we believe that the combination bounding and the failure coupling techniques should not be misunderstood or misrepresented as predictions of actual system failure behavior since this has not been reflected in actual operating experience at current generation reactors.

2. Page 74 - The probability of a very large pipe break which can cause the RCP speed to increase would be lower than the 10^{-4} for break 6" and greater.

3. Any data supporting the common mode approaches discussed would be helpful as would guidance for when tight, moderate, or loose coupling should be assumed.

APPENDIX V

1. The overwhelming conclusion is that both PWR's and BWR's have done an acceptable job in designing for the large LOCA - the DBA.

2. On page 53, a point is made that structural core and internals failures can lead to melt. These "conclusions" imply inadequacy of the ASME codes. Is this the intent?

3. A potential failure identified as ECF is given as an "adder" to the sequences. These affect the effectiveness of the ECCS. With the unjustified judgement span of 10^{-2} to 10^{-5}, implies that impared effectiveness of ECCS can dominate the core melt probabilities more than will a failure of ECCS identified in the trees. It is our opinion that this type of conjecture is out of place in WASH-1400 or any other technical treatment of ECCS. The ECF conclusion on page 54 needs to be justified.

4. The discussion of the effects of loss of off site power (page 59) is particularly confusing, when coupled with the P_1 for the failure of the RPS. (These values for P_1 are higher than the W values for ATWT).

5. Page 62 - Item Q - 10^{-2} seems adequate for value chatter. Failure to close at all seems non-mechanistic.

6. Page 76-78 - Westinghouse believes the probability of the double check valve failure has been grossly exaggerated. First, we estimate the chance that a check valve fails to open or close to be 1×10^{-5} per cycle. With the subject check valves being cycled only once per year the annual chance that it sticks open (i.e. closing on particle or hinge failure on closure) is $\lambda_1 = 10^{-5}$/year. We also estimate the chance of check valve rupture to be $\lambda_2 = 10^{-5}$/year. With these rates $\overline{Q}_{sum}$ averaged over the plant life is $3\lambda_1\lambda_2 t = 2.4 \times 10^{-8}$/year without testing for check valve closure.

In the discussion on page 78, the benefits of a testing program are not adequately accounted for by the "program" outlined. With the annual test for closure the possibility of a stuck open check valve is eliminated and $\overline{Q}_{sum} = 3(\lambda_2)^2$. Westinghouse agrees that such test would reduce the unavailability due to a stuck open valve and have recommended that utilities annually check these valves for closure.

7. Page 74 - Since the releases for the various categories were chosen to be conservative for the sequences they represent we don't see the need for adjusting the release category probabilities in the manner described.

8. Page 66 - Both P_3 and P_5 could be lower if some operator credit for correcting diesel generator faults were assumed.

APPENDIX VI

1. The atmospheric dispersion model used to predict the dilution of radioactive materials released from the plant following the accident is described as the standard Gaussian Plume Model from "Meteorology and Atomic Energy, 1968" in the Reactor Safety Study (Page 108). However, the model presented in Appendix VI of the study (Page 15) does not describe a Gaussian Plume since it does not take into account variation in cloud concentration as a function of lateral distance (See Figure 1 of RSS/W Comparison). Appendix VI describes crosswind integration.

2. The probability of death due to whole body exposures given in the Reactor Safety Study varies linearly from 0 to 1 for doses between 133 and 400 rem respectively. Doses over 400 rem are assumed to be fatal in all cases (Page 112). However, the Appendix VI to the study (Page 39) states that a linear approximation is used with zero deaths at 200 rem and 100% fatalities at 600 rem was used in the study.

3. There appears to be a discrepancy between the dose/health effects information presented in the Reactor Safety Study (Page 112) and that given in Appendix VI (Section 6.6). The discussion on Page 112 indicates that doses were broken up into three independent categories:

A)	Acute Deaths	133 to 400 rem
B)	Non-Fatal Illness	100 to 200 rem
C)	Long-Term Effects	< 100 rem

This is misleading in that long-term effects are also a result of doses that can potentially cause acute death, but by probilistic analysis do not, and non-fatal illnesses. Thus, long-term effects should be classified as less than 400 rem in all cases in which acute death is not predicted.

Additionally, no mention of this grouping is evident in Appendix VI (Section 6.6).

4. The use of a building wake factor only when the average wind speed is greater than 1.0 meter per second (Appendix VI, Page 15) appears questionable. In conservative analyses, used in the licensing of nuclear power plants, the USAEC employs a building wake factor at wind speeds much less than 1 meter per second. Additionally, the use of unrestricted credit for dilution in the building wake may be questioned. Based on experiments by the Environmental Science Services Administration at the EBR-II reactor complex in Idaho, the dispersion credit for building wake effects may be limited to a factor of 3 to 4 (i.e., $\chi/Q_{\text{no building wake}}/\chi/Q_{\text{building wake}} \leq 4$).

5. The evacuation model used in the Reactor Safety Study (Page 31, Appendix VI) may tend to overestimate evacuation rates at high density areas. The main text of the report states that the evacuation rate, which is equivalent to an evacuation half life of 2 hours, is based on an EPA study which recommends this evacuation rate for average population densities of 300 people per square mile. For these population cases analyzed in which the population near the reactor exceeds this density, it may be appropriate to use a lesser evacuation efficiency. Ideally, the evacuation rate should be a function of population density. While we realize that differences in evacuation rate are treated in the sensitivity study (Table VI-21), it would appear to be an oversimplification of evacuation considerations.

6. Many nuclear power facilities have recreational facilities developed in the immediate vicinity to enhance public acceptance of the power plant. While the presence of many thousands of additional persons near the plant conjunction with their short residence time may not affect the average results of the study, they should be included in the estimates concerning the peak consequences. Also, the unique evacuation considerations of this population segment should be taken into account.

7. It is not apparent from the report or its appendices as to the rationale for using 500 miles as the cutoff for dose considerations. It can be implied from the material presented that considerations beyond 500 miles do not change the results; however, this is not stated.

8. The Reactor Safety Study does not consider fatalities due to thyroid cancer. Although the probability of this occurrence is very small, it should be included in order to present a complete story.

9. In light of the arguments raised, it is somewhat surprising that the potential effects on the fetus of pregnant women in the exposed population is not discussed. Although it is recognized that no readily applicable data exists in this area, some qualitative discussion on this matter would be appropriate as a minimum. The draft "Swedish Urban Siting Report" prepared by the Swedish Urban Siting Committee, which covers the same basic concept as the AEC Reactor Safety Study but in less detail, has considered doses to pregnant women. The report assumed that abortions are recommended for cases in which the total body dose to a pregnant woman is greater than 10 rem.

10. The Reactor Safety Study uses an average breathing rate of 20 m^3/day (2.2×10^{-4} m^3/sec) over the entire exposure period. Since persons will generally be more active during the period of cloud passage (i.e. exposure) due to evacuation proceedings, etc. it may be more appropriate to use a breathing rate of 3.47×10^{-4} m^3/sec as used in licensing analyses.

11. It is not clear from the text of Appendix VI as to the exact cause of the fatalities. The text states that the major contributor to fatalities is the inhalation pathway with radioiodine being the critical isotope. Additionally, all fatalities appear to be a function of the whole body dose to the exposed persons near the plant. However, it is a generally recognized conception that

inhaled radioiodine contributes mainly to a thyroid dose and has a very small contribution to the whole body dose a person receives The whole radionuclide concentration to dose concept by various exposure pathways should be more fully developed in this section. We would suggest that this be accomplished in tabular form which describes the whole body dose contribution to a typical individual at some reference point from the plant as a function of isotope and dose pathway. The table would necessitate the assumptions of some specified meteorological conditions, that the individual is unaffected by evacuation, and a reference radioactivity release case.

12 The last paragraph on page 53 should follow line 3 page 54.

COMMENTS ON APPENDIX X
OF WASH-1400

Westinghouse considers that WASH-1400 is a thorough and comprehensive report on the design adequacy of nuclear power plants and concurs with the sample of nuclear power plant types, components and systems selected for the assessment of the design adequacy. Further, it appears that a judicious choice was made in the selection of the events prone to cause common-mode failures. However, Westinghouse believes that the report fails to confirm design adequacy for several items mainly due to lack of sufficient information and/or due to misinterpretations of available information. Hence, Westinghouse offers the following comments to clarify inaccuracies found in the report.

GENERAL COMMENTS

1. Westinghouse believes that the statement on page 10 on the resolution of earthquake components is incorrect. A total response of 2a will not occur since earthquakes X and Y, even when they are in phase, will occur along X and Y directions, respectively. Therefore, when they are in phase, the resultant will be $\sqrt{2}a$. However, when they are out of phase, then only 1.0a will result.

2. The discussions and conclusions about Table 14 indicate that coupling of the reactor coolant loop and concrete building is required. Westinghouse believes that this is incorrect. Since the loop frequencies are 5.12 and 8.31, respectively, as obtained by Westinghouse and Stone and Webster, these frequencies are below the concrete building frequency (5.3) in one instance, and above in the other. This means when coupled, the loop frequency will be either below 5.12 or above 8.31. As a result of the straight line response spectrum used, responses will not be changed at all. Therefore, coupling should not have any undue effect.

3. Westinghouse believes that the statement (page 21) that "a component is assumed to be subject to failure under the DBA until it is proven functional" is inaccurate. The report should state that a component is assumed to be functional if it meets all established codes and regulations as evidenced by the SAR.

SPECIFIC COMMENTS

The following comments are in direct response to conclusions reached on various parts of Westinghouse equipments.

1. Reactor Coolant Pump Nozzles (Section 6.3.2.3)

   On page 8? of WASH-1400, Appendix X, it is stated that for both nozzles, Bijlaard's method of analysis is of doubtful value for the computation of the stresses in the pump casing wall at the junction with the nozzle, since the conditions for valid application of Bijlaard's method are not present.

   Westinghouse believes that comments on the RCP nozzles are considered valid with the two following exceptions.

   a. Bijlaard's method of analysis was not used in the suction nozzle evaluation.
   b. Stresses in the casing wall for faulted conditions are contained in the report, i.e., this was the purpose of the Bijlaard analysis at the discharge nozzle.

   The comments on the limitations of the Bijlaard's method are theoretically correct, however, as in most real engineering problems some approximations must be made to arrive at a solution. In this instance, the non-uniform wall thickness of the casing was assumed to be uniform with a thickness equal to that of the casing at the centerline of the discharge nozzle. This approach is considered conservative since it does not take advantage of the local reinforcement in the nozzle area. Also, the thick flange at the end of the casing and mating attachments are believed to simulate a long cylinder.

The nozzle diameter does exceed 1/3 the diameter of the casing as stated and results in a β factor of 0.582 which is beyond the range of the Bijlaard curves. The values used in the analysis from the curves however were taken at $\beta = 0.5$ and are considered conservative because the slope of the curve in the area of $\beta = 0.5$ is either flat or decreasing.

As a further check on the validity of the above approximation, the casing has been approximated as a sphere with a radius equal to twice that of the casing to simulate membrane action. Using this approximation, the Bijlaard report may be used without extrapolating the data from the report. The results for one load case comparison show that the maximum stress occurs at the same location and differs by only 1%.

Use of the Bijlaard method and the above approximations, which make the method feasible, show that the design is adequate for its intended use. Further, finite element evaluations will be made which more closely approximate the true geometry, loadings and boundary conditions. These additional analyses are considered as only back-up to existing analyses and are not required to establish design adequacy.

2. Pipe Whip Restraints (Section 6.3.2.4)

The commentary on page 90 states that the thrust coefficient of 1.25 in the formulae for P is appropriate for the main steam line but is not sufficiently high for the feedwater line. A coefficient of 1.9 should have been used.

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

For saturated water or steam discharge through an idealized no loss nozzle, a thrust value of ~ 1.25 pA is predicted based on conservation relations and thermodynamic considerations.

Real configurations encountered however have losses due to contraction, expansion, direction change and friction. For the case of non-flashing discharge, the thrust coefficient can be expressed as:

$$\frac{p}{g} v^2 = \frac{2p}{1+K_T}$$

where, K_T is the sum of the geometric and friction head loss coefficients excluding the exit loss.

In the case of back flow from a steam generator through the feedring and out through a break in a feedline, the geometric and friction head losses are substantial. For the Surry steam generator, the flow first abruptly contracts into 240 holes, 0.75 in. diameter in the feedring and then expands into the feedring cross section. The head loss (in terms of feedline velocity) is estimated to be 0.78 for these holes. The two ends of the circular feedring join in a Tee junction which connects to the feedline nozzle. The head loss for collection at the Tee is estimated to be 1.11. Outside the steam generator the first long radius 90° elbow is estimated to have a head loss of 0.26. The total head loss is K_T = 2.15. The resultant thrust coefficient for non-flashing water is 0.64. For saturated water, Figure 8 of Reference 1 indicates a thrust coefficient of about 0.85 for f^L/D = 2.15. In the case of feedline break, the initial blowdown would be subcooled but flashing water should yield a thrust coefficient between these extremes.

It is concluded that a thrust coefficient of 1.25 x pA is conservative for a feedline break.

Reference 1: Fluid Reaction and Impingement Loads, F. J. Moody

3. LHSIS Pump Shaft (Section 6.3.3.2)

WASH-1400 concludes that it is not certain that LHSIS pump can continue to function during and after impeller deflection of 1.15 in. and that no tests or analyses were performed to provide this assurance.

Calculations show that a momentary interference between the rotating and stationary elements would not be detrimental for a static condition and an interference would not exist at all during the actual operation of the pump.

This conclusion, coupled with the calculated stresses which are shown to be below the allowable stresses, assures design adequacy of the pumps.

4. Radiation Effects on Pumps (Section 6.3.3.2 and 6.3.5.1)

Westinghouse believes that the radiation resistance of pump internals, questioned in Section 6.3.3.2 and 6.3.5.1 has been demonstrated for the following reasons.

a. Table 1, page 13 of WASH-1400, Appendix X gives extremely conservative integrated radiation dosage with the actual expected to be about 100 times less than the maximum design requirement of 1.5×10^8. As discussed below pump materials have been evaluated after gamma exposures of 1.1×10^8 and 1×10^9 rads. Hence, for the materials of interest, Westinghouse believes that adequate test data to confirm adequacy at the expected exposure exists.

b. The internal components of the low heat safety injection pumps are constructed primarily of stainless steels. No deleterious effects on stainless due to gamma radiation during a LOCA type accident are expected.

c. The materials other than steel used in internal parts are discussed below:

(1) Graphitar 14 is used for internal shaft bushings. This material, a carbon graphite substance, is relatively inert and has been used extensively in pumps for radioactive fluids in nuclear installations, including Hanford, Naval Reactors Facility, Oak Ridge National Laboratory and fuel reprocessing plants. The satisfactory performance in these applications is supported by results in APEX 357, "Estimated Radiation Stability of Aircraft Components" (G. E. Atomic Products Division) which reports data on Graphitar 39, a similar material. No significant damage was reported at approximately 1×10^9 rads.

(2) An EPT elastomer rubber formulation is used for making the bellows and "O" rings in the Crane Packing Company mechanical seal under conditions postulating the accident conditions and including tests on bellows to 1.1×10^8 rads dosage were conducted to qualify the mechanical seal for this application. The results were satisfactory and are reported in Crane Packing Company Bulletin No. 3472.

5. Accumulator Tank Nozzle (Section 6.3.4.1)

It is stated on page 108 of WASH-1400 that there exists an inconsistency in the results, and the results of the analysis and evaluation performed by the supplier are questionable. The inconsistency pointed out in the report is that the total membrane stress due to external load on Tank No. 3 was less than the corresponding value for Tank 2, even though the pipe loads on Tank No. 3 were greater than those on Tank No. 2.

Westinghouse believes that an incorrect comparison of Primary Local Membrane stresses (σ_L) was made in the report for Loop No. 2 and Loop No. 3 accumulators. A detailed comparison of the loadings and the resulting stresses is presented in the Table I. This table shows that there is no inconsistency in the results.

6. Sensors and Logic Cabinets (Section 6.3.8)

The report states that the qualification of the sensors and logic cabinets could not be evaluated for seismic and steam environmental exposures with the information available.

Westinghouse believes that the additional information provided below shows that Westinghouse did conduct seismic qualification tests at substantially higher input levels than that contained in Reference 26 of WASH-1400, and the protection equipment was exposed to various environmental conditions.

Based on the above, it is concluded that the PWR components are in fact adequately qualified.

TABLE I

Accumulator Tank Nozzle

Stresses in Head at Nozzle				
No.	Origin and Type of Stress	Classification	Loop No. 2	Loop No. 3
(1)	Int. pressure - membrane	σ_m	16780 psi	16780 psi
(2)	Allowable stress per code case 1607 - general membrane	2.0S	34580	34580
(3)	Ext. piping loads - membrane	σ_{L1}	2370	10410
(4)	Int. pressure and ext. loads local membrane (1) + (3)	σ_L	19150	27190
(5)	Any mechanical load-bending (Not applicable at discontinuity)	σ_b	0	0
(6)	Int. pressure and ext. loads - total stress (4) + (5)	$\sigma_L + \sigma_b$	19150	27190
(7)	Allowable stress per code case 1607	2.4S	41496	41496
(8)	Any mechanical load-bending (at discontinuity)	0	13300	Not tabulated since present criteria per code case 1607 does not call for the evaluation of this stress.
(9)	Int. pressure + ext. loads - total (6) + (8)	$\sigma_L + \sigma_b Q$	32450	
(10)	Allowable stress		Code case 1607 does not call for the evaluation of this stress.	

SENSORS AND LOGIC CABINETS
(Environmental)

The equipment samples which were environmentally tested were not subjected to simulated normal aging prior to the environmental tests, as is a current requirement. However, the tested components are parts of repairable systems which are required to be tested at one month intervals during operation with the reactor at power. If normal aging should cause any degradation of component performanc. , such degradation would be detected by these periodic tests and the component would be replaced. It should also be noted that during the qualification testing the test samples were subjected to the LOCA environment for a period much longer than the period much longer than the period of required operation following the postulated LOCA. This excess time at LOCA conditions is equivalent to a much longer period of normal aging.

Instrumentation set point drift under normal plant operating conditions is a condition which would be immediately detectable by the required periodic testing. Plant experience would rule out the use of families of instrumentation with a history of excessive drift. There is, of course, the possibility of individual instrument drift in any system. Due to fail safe philosophy employed in safety system design, the probability that such drift would be in the safe direction is somewhat greater than 0.5. In any case, system redundancy is such that excessive drift in a number of instruments could occur during the interval between tests without compromising the system safety functions.

In the qualification test program, the equipment was exposed to various environmental conditions sequentially rather than to combinations of conditions, e.g. seismic simulation in combination with steam.

Based on the above considerations, the protection system is expected, with a high degree of confidence, to perform its safety functions under accident conditions.

SENSORS AND LOGIC CABINETS
(Seismic)

The concept of "type" testing (testing of a typical assembly that contains typical components or performs a particular type of function) has been accepted by the USAEC and is also acceptable in accordance with the requirements of the Draft, Rev. 4 of IEEE 344 - 1974, which reflects the latest technology for seismic qualification of electrical and I&C equipment.

Westinghouse conducted seismic qualification tests at substantially higher input levels than that contained in Reference 26 of WASH-1400. In WCAP-7821[1], the equipment was subjected to input accelerations of 1.5 g maximum and for the testing conducted for WCAP-8021 inputs of 2.0 g maximum were used.[2] The WCAP-7817 (Reference 26, WASH-1400) testing was used to a certain extent to finalize the testing criteria for the subsequent higher g level input tests reported in WCAP-7821 and WCAP-8021. These latter two test programs were conducted in accordance with the criteria stated in WCAP-7817 and again repeated in WCAP-7821 and 8021.

In the three test programs (using increasingly higher inputs) in general, the same electrical equipment was tested. Therefore, this equipment which was tested three times was subjected to a fatigue environment far in excess of actual earthquake requirements.

In recognition of this fact, the statement that "failures or malfunction caused by wear or fatigue resulting from imposing more than 100 cycles on the unit tested does not constitue failure of the seismic test" was added to the criteria to partially compensate for the overly conservative fatigue testing. In practice this provision was used only when the malfunction could be clearly related to the fatigue over testing. This provision was used for the following typical obvious fatigue failures which met the stated criteria requirements:

1. Static Inverter

   a. After third test, the insulation for one wire wore through
   b. After third test, some self-taping sheet metal screws on one side panel came loose.
   c. After third test, bolting hardware for a capacitor bracket came loose.

2. WCAP-7821, Supplement #2, Section 4.1.1 - Static Inverter

   a. Base weld crack, after second test

In general, if a structural failure or malfunction took place, the equipment design was modified and the equipment was then retested to ensure compliance with the seismic criteria.

In regard to the justification of the single frequency, and axis tests performed. WCAP-8373[3] and WCAP-7558[4] demonstrate that the testing performed in accordance with WCAP-7817, 7821 and 8021 is a severe and conservative representation of possible seismic conditions for the equipment. Also, WCAP-8373 presents the seismic qualification levels for the various types of equipment tested. The qualification levels are related to actual floor motions at the equipment installed positions.

References

1. WCAP-7821, "Seismic Testing of Electrical and Control Equipment (High Seismic Plants)", by L. M. Potochnik, December 1971, and Supplements 1 through 3.

2. WCAP-8021, "Seismic Testing of Electrical and Control Equipment (PG&E Plants), by L. M. Potochnik, May 1973.

3. WCAP-8373, "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May, 1974", by E. G. Fisher and S. J. Jarecki, August 1974.

4. WCAP-7558, "Seismic Vibration Testing with Sine Beats", by A. Morrone, October, 1971.

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

[redacted]

[redacted]
One IBM Plaza
Suite 4600
Chicago, Illinois

IN RESPONSE REFER
TO FOIA-77-A-16

Dear Mr. Osann:

This is in response to your letter to me of June 4, 1977, in which you appealed the May 3, 1977, and June 1, 1977, initial denial of your Freedom of Information Act request of March 28, 1977, for documents relating to the abnormal occurrences or reactor malfunctions involving the pressure suppression system and/or containment of the Gundremmingen and the Wuergassen Nuclear Power Plants in West Germany, including communications to or from Northern Indiana Public Service Company. [redacted] Based on this review I have [redacted] cluded that the [redacted]. Accordingly, a copy of each document is enclosed.

The documents listed in [redacted] for review in accordance with 10 CFR 9.5(c). The Department of [redacted] s and a copy of each document is enclosed. [redacted] You will be notified as soon as that review is completed.

After careful consideration, I have determined that [redacted] under exemption (1) of the Freedom of Information Act (5 U.S.C. 552(b)(1)) and 10 CFR 9.5(a)(1) of the Commission's regulations and that disclosure of these documents would be contrary to the public interest. [redacted]

Each of the documents referred to above is properly classified under the criteria of Executive Order No. 11652 of March 8, 1972. The Executive Order specifies that information which requires protection in the interest of the national security must be classified. The information at issue is of foreign origin and was received by the Nuclear Regulatory Commission in confidence only after express assurances were made that such information would be protected from disclosure. Accordingly, the release of this information would have a significant adverse impact on the "conduct of our foreign relations," and therefore is properly classified as national security information. On this basis, the aforementioned documents have been assigned a security classification.

Additionally, I have reviewed the record in this case in light of the following considerations and have further determined that some of the documents denied above on exemption (1) grounds and [redacted] of the Freedom of Information Act (5 U.S.C. 552(b)(4)) and 10 CFR 9.5(a)(4) of the Commission's regulations. [redacted] A copy of document C-3 with exempt portions deleted is enclosed.

The fourth exemption of the Freedom of Information Act excludes from mandatory disclosure matters that are "trade secrets and commercial or financial information obtained from a person and privileged or confidential," 5 U.S.C. 552(b)(4). The test for determining whether commercial or financial matters are "confidential" within the meaning of the exemption is dependent upon whether disclosure of the information is likely to have either of two effects: (1) impair the Government's ability to obtain necessary information in the future; or (2) cause substantial harm to the competitive position of the person from whom the information was obtained. National Parks and Conservation Association v. Morton, 498 F.2d 765, 770 (D.C. Cir 1974). Further, the exemption may be invoked even though the Government itself has no interest in keeping the information secret. Id.

The documents at issue were obtained by the Nuclear Regulatory Commission (NRC) when it sought to obtain information regarding a reactor incident in a foreign country in order to determine whether the incident would have an impact on the operation of domestic nuclear power plants. Inasmuch as the NRC was not otherwise entitled under the law to obtain such information, and since the holder of such information was unwilling or unable to otherwise disclose the informat on to the NRC, the NRC agreed that it would treat the information in confidence should the holder divulge the information. Under these conditions the information was disclosed to the NRC.

As I am sure you appreciate, it is imperative that the NRC be able to obtain all available reactor safety information in order to properly fulfull its statutory mandate to protect the public health and safety. When reactor safety information is of foreign origin, the NRC cannot, under most circumstances, obtain the information other than by complying with the conditions under which it is supplied to NRC. Therefore, it is essential that NRC protect foreign information given in confidence from pub ic disclosure or risk the loss of this important source of r.actor safety information.

In conjunction with the above, it should be noted that approximately 50% of the operating nuclear power reactors (many of which were designed by U.S. vendors) are located outside the United States, and thus approximately 50% of the operating reactor safety information is generated outside the United States. When viewed in this light, I am sure you can appreciate why it is vital to the proper functioning of NRC that our flow of foreign safety information not be jeopardized by the disclosure of information supplied in confidence.

On appeal you request that each withheld document

> "be identified as to author or addressor, date, person or persons to whom directed, identification of subject matter, and specific ground for withholding."

[redacted]. Release of such particulars in the face of a pledge of confidentiality would reasonably be expected to cause damage to the foreign relations of the United States, and to impair the ability of the NRC to obtain necessary reactor safety information in the future.

[redacted]. As set forth in the Freedom of Information Act (5 U.S.C. 552(a)(4)(B)), [redacted] of the United States in either the district in which you reside, have your principal place of business, or in the District of Columbia.

Sincerely,

Lee V. Gossick
Executive Director
for Operations

Distribution:
LVGossick
WJDircks
TARehm
DJDonoghue
JMFelton
JWMaynard
DCDambly — Gertter #2066
SFEilperin — HFaulkner
SPMurray — WGDooly
JDLafleur
EGCase

See attached for previous concurrences

OFFICE	ELD	ELD	IP	NRR	OGC	EDO
SURNAME	JWMaynard	JPMurray	JDLafleur	EGCase	SEilperin	LVGossick
DATE	7/8/77	7/8/77	7/ /77	7/11/77	7/8/77	7/ /77

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APPENDIX A

1. Memorandum to the Files, dated January 19, 1977, Subject: Information on Gundremmingen Incident.

2. Implication on Domestic Facilities.

3. KRB Event Knowns.

APPENDIX B

1. State cable 017936, 1/24/77, Subject: Incident at Gundremmingen Nuclear Power Station.

2. State cable Brussel 1748, March 29, 1973, Subject: Wurgassen Nuclear Power Plant - Steam Line Leak.

3. State cable Brussels 10450, November 21, 1975, Subject: Gundremmingen Nuclear Power Station.

4. State cable, Bonn 01869, February 1, 1977, Subject: Gundremmingen Nuclear Power Plant Incident.

5. State cable, Bonn 00951, January 18, 1977, Subject: Nuclear Incident in German Gundremmingen Nuclear Power Station.

APPENDIX C

1. Memorandum for V. Stello from D. Eisenhut, undated, Subject: Summary of Conference Call.

2. Preliminary Evaluation - KRB 13 January 1977 Loss-of-Load

3. KRB Incident of January 13, 1977

FROM:
Edward W. Osann
Chicago, Ill.

TO:
Exec. Dir.

ACTION CONTROL	DATES	CONTROL NO.
COMPL DEADLINE	7/11/77	2066
ACKNOWLEDGMENT		DATE OF DOCUMENT
INTERIM REPLY		6/4/77
FINAL REPLY		PREPARE FOR SIGNATURE OF: ☐ CHAIRMAN ☒ EXECUTIVE DIRECTOR OTHER:
FILE LOCATION		

DESCRIPTION ☒ LETTER ☐ MEMO ☐ REPORT ☐ OTHER

Appeal from Initial FOIA Decisions (FOIA-77-61) a req for records relating to abnormal occurrences or reactor malfunctions involving the pressure suppression system and/or containment of the Gundremingen & Wurgassen nuclear power plants in Germany

SPECIAL INSTRUCTIONS OR REMARKS EDO-1745

Draft response to EDO/ELD no later than 7/1/77

APPEAL OF INITIAL FOIA DECISION

FOIA-77-A-16

SECY

CLASSIFIED DATA

DOCUMENT/COPY NO.			
NUMBER OF PAGES		CLASSIFICATION	
POSTAL REGISTRY NO.		CATEGORY ☐ NSI ☐ RD ☐ FRD	

ASSIGNED TO:	DATE	INFORMATION ROUTING	
Dooly	6/10/77	Gossick	Dambly
		Dircks	Eilperin
		Rehm	
		Donoghue	
		Felton	
		Maynard	

LEGAL REVIEW ☐ FINAL ☐ COPY

ASSIGNED TO:	DATE

NO LEGAL OBJECTIONS
NOTIFY:
☐ EDO ADMIN & CORRES BR
EXT.
COMMENTS, NOTIFY:
EXT.

JCAE NOTIFICATION RECOMMENDED: ☐ YES ☐ NO

NRC FORM 232 (11-75)

EXECUTIVE DIRECTOR FOR OPERATIONS

PRINCIPAL CORRESPONDENCE CONTROL - DO NOT REMOVE THIS COPY

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or mention abnormal occ... pressure suppression system and/or containment of ... and the Würgassen Nuclear Power Plants in West Germany, including communications to or from Northern Indiana Public Service Company. The records sought are all of those for the period from January 1, 1970 to and including March 28, 1977.

It is urged that the initial decisions be reversed and the withheld records be produced to the undersigned without further delay. In the event, however, that the Executive Director for Operations should sustain the withholding of one or more of said records, it is requested that each withheld document be identified as to author or addressor, date, person or persons to whom directed, identification of subject matter, and specific ground for withholding.

Very truly yours,

Edward W. Osann, Jr.

Edward W. Osann, Jr.

One of the Attorneys for the Izaak Walton League of America, Inc., Concerned Citizens Against Bailly Nuclear Site, Businessmen for the Public Interest, Inc James E. Newman, Mildred Warner and George Hanks.

EWO/mk

cc: Robert J. Vollen, Esq.
Robert L. Graham, Esq.

APPENDIX A

1. Memorandum to the Files, dated January 19, 1977, Subject: Information on Gundremmingen Incident.

2. Implication on Domestic Facilities.

3. KRB Event - Knowns.

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

Exh #5

January 19, 1977

MEMORANDUM TO THE FILES

SUBJECT: INFORMATION ON GUNDREMMINGEN INCIDENT

I talked by telephone with Helmut Schnurer, in the Nuclear Safety Office of the FRG Ministry of the Interior (FRG-BMI), on January 19, 1977, in follow-up of our talk of January 18 on the subject of the incident of January 13, 1977 at Gundremmingen.

A man from FRG-BMI had returned from Gundremmingen late January 19. A report will be made to the Minister (of BMI) on January 20, 1977. It is expected that a public announcement will be issued, also on January 20. Schnurer will telegraph the text of this announcement to me.

No public disclosure is to be made until after announcement by the Minister. Only what was shown in the January 18 telegram as already announced can be released.

The following is the information brought back from the site on January 19:

NOT YET ANNOUNCED

1. As NRC had heard from others, there was some trouble with a pressure relief valve.

2. Of the 13 or 14 valves that opened as designed, one valve was bowed away so that it could not close. This led to continued pressure release into containment. The one valve failed to close because it was "rotated about 20° from its usual position." (Does not know whether it was internally or externally rotated.)

3. The piping is tight--valve did not fly off.

4. All other associated systems worked as intended.

5. Much primary water vented into the containment. 460 cubic meters collected in containment from both primary system and containment spray. Water now has been pumped to the feedwater storage tanks. None was released to environment.

POOR ORIGINAL

NOT FOR RELEASE WITHOUT PERMISSION OF THE GERMAN MINISTRY OF INTERIOR

This info released by FRG BMI 2/1/77

cc: J. R. Shea
V. Stello
L. Sleggers
H. Faulkner

Joseph D. Lafleur, Jr., Deputy Director
Office of International Programs

JDL
4/26

Lycera Inspection 3/1977 Exhibit 5 | A-1

IMPLICATION ON DOMESTIC FACILITIES

- NRC WORKING CLOSELY WITH GE
- GE HAS TWO PHASE EFFORT
  - REVIEW OF BWRs FOR SIMILARITIES
  - MODELING STUDIES TO UNDERSTAND CAUSE

- SURVEY OF U.S. BWR DESIGNS
- NO U.S. PLANTS WITH GERMAN SRV
  - NONE HAVE UNIQUE HEADER DESIGN
  - MOST HAVE SRV ON STEAMLINES
  - HB/NMP HAVE SRV ON HEAD
  - DI/BR HAVE SRV ON STEAM DRUM

- FEEDWATER TRIPS
  - ALL CP/OL
  - MOST ORs
  - DO NOT HAVE TRIP

- U.S. EXPERIENCE
  - 4 INCIDENTS
  - NO FAILURES

KRB EVENT

KNOWNS

- LOSS OF 220KV LOAD
- TURBINE CONTROLLER MALFUNCTION
- 380V BUS DE-ENERGIZED
- MULTIPLE SSV OPENINGS
- VALVE ATTACHMENT FAILURE
- SAFETY SYSTEM RESPONSE AS EXPECTED
- ECCS OPERATION CORRECT
- NO RADIATION RELEASES
- NUCLEAR INSTRUMENTATION FUNCTIONAL POST EVENT
- SOME ELECTRICAL EQUIPMENT WATER DAMAGE

BASED ON INFORMATION RECEIVED IN
CONFIDENCE FROM A FOREIGN SOURCE

APPENDIX B

1. State cable 017936, 1/24/77, Subject: Incident at Gundremmingen Nuclear Power Station.

2. State cable Brussel 1748, March 29, 1973, Subject: Wurgassen Nuclear Power Plant - Steam Line Leak.

3. State cable Brussels 10450, November 21, 1975, Subject: Gundremmingen Nuclear Power Station.

4. State cable, Bonn 01869, February 1, 1977, Subject: Gundremmingen Nuclear Power Plant Incident.

5. State cable, Bonn 00951, January 18, 1977, Subject: Nuclear Incident in German Gundremmingen Nuclear Power Station.

Department of State (1) TELEGRAM

ORIGIN NRC-07

INFO OCT-01 AS-01 EUR-12 IO-13 ISO-00 OES-06 ACDA-07 FEA-01
CIAE-00 INR-07 L-03 NSAE-00 NSC-05 EB-08 DODE-00
ERDA-05 PM-04 SS-15 /095 R

DRAFTED BY U.S. NRC: J. D. LAFLEUR:SMJ
APPROVED BY OBHOYLE OES/NET/IM
EUR/RPE: A. D. SENS
ACDA: D. OYSTER

------------------270206Z 057897 /62

P R 262250Z JAN 77
FM SECSTATE WASHDC
TO AMEMBASSY BONN PRIORITY
INFO USMISSION EC BRUSSELS
USMISSION IAEA VIENNA

LIMITED OFFICIAL USE STATE 017936

E.O. 11652: N/A

TAGS: TECH, GW

SUBJECT: INCIDENT AT GUNDREMMINGEN NUCLEAR POWER STATION

REFERENCE: A) BONN 00951; B) TELEGRAM 17 JAN: NRC TO SCHNURER - INFO TO EMBASSY

1. APPRECIATE INFORMATION REGARDING GUNDREMMINGEN INCIDENT PROVIDED IN REFTEL A. FRG MINISTRY OF INTERIOR ALSO PROVIDED A DESCRIPTION OF THE INCIDENT TO US AND WILL PROVIDE A COPY OF FRG PRESS RELEASE WHEN AVAILABLE.

2. WE HAVE RECEIVED CONFLICTING STORIES REGARDING THE MALFUNCTION OF ONE OF THE REACTOR SAFETY VALVES. FRG STATES THAT ONE VALVE WAS DEFORMED TO THE EXTENT THAT VALVE WOULD NOT SUBSEQUENTLY RESEAT. OTHER SOURCES INDICATE THAT ONE SAFETY VALVE WAS BROKEN OFF (PHYSICALLY SEPARATED) FROM THE PIPING. FRG-BMI (SCHNURER) HAS PROMISED TO COMMUNICATE WITH NRC FURTHER ON THIS MATTER, AND WE PREFER NO ADDITIONAL CONTACTS BY EMBASSY WITH FRG AT THIS TIME. HOWEVER, WE WOULD APPRECIATE RECEIVING ANY INFORMATION YOU OBTAIN FROM MEDIA REPORTS.
VANCE

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1973 MAR 29 PM 2 51

U.S. ATOMIC ENERGY COMM.
MAIL & RECORDS SECTION

O'Leary
Hendrie 3/30/73

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FOR BECK REG
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ZNR UUUUU ZZH
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FM USMISSION EC BRUSSELS
TO RHEGAAA/AEC GERMANTOWN 563
INFO RUFHOL/AMEMBASSY BONN 2552
BT
UNCLAS EC BRUSSELS 1748

AEC FOR BECK, DR AND BURKE, IP

E.O. 11652: N/A
TAGS: TECH, [illegible]
SUBJ: WUERGASSEN NUCLEAR POWER PLANT- STEAM LINE LEAK

ie donot reveal source

REF: BRADLEY-[illegible] TELECON MAR 27

1. MISSION HAS LEARNED FROM EC SOURCE WHO HAD VISITED PLANT [illegible] THAT A LEAK AT [illegible] OF THE PRESSURE RELIEF LINE CONNECTION TO THE MAIN STEAM LINE WAS DISCOVERED ON FEB. 16 IN THE COURSE OF INSPECTING THIS 640 MWE AEG-BUILT BWR PLANT FOR OTHER REASONS. ACCUMULATION OF ABOUT 400 LITERS OF WATER PER DAY IN THE BUILDING VENTILATING EXHAUST SYSTEM AND A HIGH HYDROGEN CONTENT IN THE AIR WITHIN THE CONTAINMENT BUILDING HAD SUGGESTED POSSIBLE STEAM LEAKAGE FOR SOME TIME.

2. THE LEAKAGE OCCURRED IN THE WELD ZONE BETWEEN THE FORGED MAIN STEAM LINE REDUCING TEE AND THE SMALLER DIAMETER PIPING LEADING TO A PRESSURE RELIEF VALVE.

3. PRELIMINARY INVESTIGATION INDICATES THE WELDED JOINT COULD HAVE BEEN OVERSTRESSED BY WATER-HAMMER EFFECTS ATTENDING PLANT CYCLING WATER FLUSHING OPERATIONS UNDER PRESSURE PRECEDING INITIAL STARTUP OF THE PLANT.

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4. ONLY BUTT WELD CONNECTIONS, PERMITTING 100 PERCENT RADIOGRAPHY, ARE BEING USED IN THE REPLACEMENT OF THE LEAKING REDUCING TEE AND THE THREE SIMILAR NON-LEAKING TEES IN ORDER TO AVOID A PRESSURE TEST OF THE ENTIRE PRIMARY SYSTEM.

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5. MISSION WILL FORWARD FURTHER DETAILS WHEN AVAILABLE.

Department of State TELEGRAM

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PAGE 01 241446Z

File Germany

41 49
ACTION NRC-07

INFO OCT 01 SSO-00 AS-01 /009 W
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O 211503Z NOV 75
FM /USMISSION/EC/BRUSSELS
TO NRC WASHINGTON IMMEDIATE
ERDA GERMANTOWN IMMEDIATE
INFO AMEMBASSY BONN IMMEDIATE

UNCLAS EC BRUSSELS 10450

NRC PASS TO L. GOSSICK, H.C. KOUTS, AND J. LAFLEUR

ERDA PASS TO N STEVERING, AIA

E.O. 11652: N/A
TAGS: TECHN ENRG, GW
SUBJECT: GUNDREMMINGEN NUCLEAR POWER STATION.

1. FOLLOWING TECHNICAL INFORMATI WHICH HAS BEEN OBTAINED INFORMALLY IS PROVIDED TO SUPPLEME AMEMBASSY BONN, LOU CABLE 19832, DATED NOV. 20.

2. INCIDENT OCCURRED AT 10:45 A.M. ON NOVEMBER 19. THE REACTOR HAD BEEN DOWN AND WAS AT HOT STANDBY: ZERO POWER, 520 DEGREES F, 24[illegible] PSI.

3. CAUSE OF INCIDENT WAS DEFECTIVE 4 INCH VALVE, WHOSE PACKING HAD BEEN LEAKING FOR SOME TIME. VALVE WAS LOCATED IN A SIX FOOT BY TWELVE FOOT BY TWELVE FOOT CELL AND WAS PART OF A PRIMARY COOLANT PURIFICATION SYSTEM LEADING TO ION EXCHANGE COLUMNS. VALVE WAS ON A BY-PASS EMINATING FROM ONE OF TWO PARALLEL LINES WHICH (JOINED TOGETHER) CONNECTED TO THE PRIMARY CIRCUIT.

4. TWO WORKERS WERE ASSIGNED TO REPLACE PACKING, THEY UNSCREWED THE PACKING HOLD DOWN DEVICE AND WERE SPRAYED WITH SUPERHEATED STEAM. ONE MAN DIED IMMEDIATELY, THE OTHER WAS SERIOUSLY BURNED.

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Department of State TELEGRAM

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EXCESSIVE CONTAINMENT HUMIDITY WAS INDICATED IN CONTROL ROOM, AND AN INVESTIGATOR WAS SENT TO THE AREA. INJURED MANS FOUND ON LADDER NEAR CELL EXIT AND WAS SENT TO HOSPITAL SPECIALIZING IN TREATMENT OF SEVERE BURN VICTIMS. HE WAS " SUPERFICIALLY" RADIOACTIVE FROM STEAM SPRAY. RADIATION LEVEL NOT STATED NOR WHETHER MEDICAL TREATMENT WAS IMPEDED BY RADIATION. WORKER DIE NEXT MORNING.

5. CONTAINMENT RADIATION LEVEL ROSE TO TWICE NORMAL BUT EXACT VALUE NOT STATED. RELEASE AT STACK WAS MINIMAL (BARELY DISTINGUISHABLE PEAK) OBVIOUSLY CONSIDERABLY BELOW MAXIMUM PERMISSIBLE. THERE WAS NO RECORDED INCREASE IN GROUND LEVEL RADIATION. REPAIR WORK WAS PERFORMED IN CELL SAME DAY AS INCIDENT. PLANT ALMOST AT FULL POWER AS OF NOON, NOVEMBER 21, 1975.

6. TWO TENTATIVE EXPLANATIONS ARE PROPOSED FOR INCIDENT.
A. THE VALVE WAS CLOSED. SEVERAL LITERS OF RESIDUAL WATER REMAINED IN THE UPPER VALVE BODY THERE WAS NO DISCHARGE LINE AND THUS NO WAY TO BLEED INTERNAL WATER. WORKERS LOOSENED HOLD DOWN DEVICE AND TAPPED VALVE BODY TO SLOWLY FLASH HOT PRESSURIZED WATER THROUGH PACKING. NOTING NO RELEASE,WORKERS REMOVED HOLD DOWN DEVICE RELEASING PACKING AND STEAM.

B. ABOVE EXPLANATION DOES NOT APPEAR TO PROVIDE FOR SUFFICIENT STEAM RELEASE TO ACCOUNT FOR CONTROL ROOM INDICATION. ALTERNATE VIEW IS THAT THE DEFECTIVE, REMOTELY CLOSED VALVE WAS NOT COMPLETELY CLOSED (NOR WAS IT HAND TIGHTENED), THUS STEAM COULD DISCHARGE THROUGH VALVE AS PACKING WAS REMOVED. IN THIS CASE, INCIDENT WAS ENDED ONLY AFTER CONTROL ROOM OPERATOR ISOLATED EFFECTED SECTION.

7. IT IS PLANNED TO ISSUE A FULL REPORT IN A MONTH OR SO.. AN INVESTIGATION IS UNDERWAY TO DETERMINE WHETHER SUFFICIENT RULES EXISTED FOR VALVE REPAIR OR WHETHER THERE MAY BE A GENERAL DEFICIENCY IN THIS AREA(FOR EXAMPLE, ABSENCE OF DISCHARGE LINE). MORRIS

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PAGE 01 BONN 01869 01 OF 02 011139Z

ACTION

ACTION NRC-07

INFO OCT-01 EUR-12 ISO-00 OES-06 FEA-01 ACDA-10 CIAE-00
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UNCLAS SECTION 01 OF 02 BONN 01869

DEPT PASS TO DR. LAFLEUR, NRC, AS QUICKLY AS POSSIBLE

E.O. 11652: N/A
TAGS: ENRG, TECH, GE
SUBJECT: GUNDREMMINGEN NUCLEAR POWER PLANT INCIDENT

REF: A. BONN [illegible] B. LAFLEUR/MCCLELLAND TELCONS

1. EMBASSY POUCHING ASAP THREE REPORTS PROVIDED BY FRG MINISTRY OF INTERIOR (BMI):

A. REPORT TO BAVARIAN PARLIAMENT BY BAVARIAN MINISTRY OF REGIONAL PLANNING AND ENVIRONMENTAL PLANNING, JANUARY 19, 1977.

B. TELEX [illegible] FROM BAVARIAN MINISTRY TO MINISTRY OF INTERIOR; AND

C. PRESS RELEASE MUNICH, JANUARY 14, 1977.

2. EMBASSY WILL INCLUDE OUR INFORMAL TRANSLATION OF 1. A. REPORT.

3. NONE OF THESE REPORTS INCLUDE TECHNICAL DATA ABOUT REPORTED VALVE FAILURES.

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Department of State TELEGRAM

4. DR. WEIL, BMI, REVIEWED WITH SCIENCE COUNSELOR THE DATA HE PROVIDED TO FAULKNER, NRC, BY TELEPHONE ON JANUARY 31. THESE DATA FOLLOW:

ALL FOURTEEN SPRING-OPERATED PRIMARY CIRCUIT PRESSURE RELIEF VALVES OPERATED, AS INDICATED BY BURST DIAPHRAGMS . PIPE CONNECTING ONE VALVE TO CIRCUIT APPARENTLY RUPTURED AT -- OR NEAR -- WELDED SEAM CONNECTING VALVE TO PIPE. ABOUT THREE-FOURTHS OF PIPE CIRCUMFERENCE IS RUPTURED, AND VALVE NOW SITS AT AN ANGLE. VERTICAL PIPE NOZZLE AT THIS POINT IS NARROWING FROM 165 MM TO 94 MM, INSIDE DIAMETER. BMI BELIEVES SEAM -- OR ADJACENT PIPE -- FAILED BECAUSE NOT ONLY STEAM BUT EXCESS WATER WAS PRESENT IN REACTOR VESSEL DUE TO OVER-FEEDING BY FEED WATER PUMPS THROUGH OPEN, REMOTELY-OPERATED VALVE WHICH COULD NOT BE CLOSED FOLLOWING 2.8 SECOND SHUT-DOWN FOR SWITCHING STATION POWER TO INDEPENDENT HIGH VOLTAGE LINE.

5. REACTOR OPERATOR HAS SUBMITTED REPORT ON INCIDENT TO BMI WHICH IS BEING STUDIED. FACTS, AS VERIFIED, WILL BE INCLUDED IN OFFICIAL BMI REPORT WHICH WILL BE FORWARDED TO NRC THROUGH EMBASSY ASAP AFTER APPROVAL BY MINISTER, PROBABLY IN FEW DAYS.

6. INVESTIGATION CURRENTLY DELAYED BY DECONTAMINATION OPERATIONS USING STEAM. ENTRY INTO CONTAINMENT WILL AGAIN BE POSSIBLE IN FEW DAYS AND INVESTIGATION WILL BE COMPLETED THEN.

7. BMI PLACES NO LIMITATION ON USE BY NRC OF DATA OUTLINED IN PARA 4, ABOVE. THESE FACTS WILL BE PUBLISHED SOON.

8. FIRST OF THREE REPORTS (SEE PARA 1. A., ABOVE)

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WAS PASSED TO EMBASSY "NOT FOR DISSEMINATION WITHOUT APPROVAL OF THE BMI OR THE NRC." HOWEVER, BMI SUBSEQUENTLY WITHDREW THIS LIMITATION - AS MARKED ON COPY POUCHED.

9. FOLLOWING ARE EXCERPTS OF REPORT MENTIONED IN PARA 1. A., ABOVE, WHICH RELATE TO OPERATION OF PRESSURE RELIEF VALVES DURING INCIDENT. NUCLEAR POWER PLANT WAS CUT OFF FROM GRID, THEN:

BEGIN INFORMAL TRANSLATION OF EXCERPTS: QUOTE ... DELAYED REACTION OF TURBINE REGULATOR RESULTED IN SUDDEN PRESSURE DROP IN PRIMARY STEAM CIRCUIT ... PRESSURE SENSOR REACTED...SHUT VALVE IN PRIMARY STEAM CIRCUIT...SWITCHED STATION POWER FROM 220 KV LINE TO

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PAGE 01 BONN 01869 02 OF 02 011144Z
ACTION NRC-07

INFO OCT-01 EUR-12 ISO-00 FEA-01 ACDA-10 CIAE-00 INR-07
IO-13 L-03 NSAE-00 NSC-05 EB-08 OES-06 DODE-00
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FM AMEMBASSY BONN
TO SECSTATE WASHDC IMMEDIATE 5210

UNCLAS SECTION 02 OF 02 BONN 01869

INDEPENDENT 110 KV LINE...STATION WITHOUT POWER FOR 0.8 SECONDS...INCLUDING FEED WATER REGULATOR EQUIPMENT. THE REMOTELY OPERATED VALVE, WHICH REGULATES FEED WATER FLOW TO THE REACTOR PRESSURE VESSEL WAS IN THE OPEN POSITION. THE FEED WATER PUMPS, WHICH WERE SWITCHED ON AGAIN, SUPPLIED WATER TO THE REACTOR PRESSURE VESSEL (18 CUBIC METERS PER MINUTE). WHEN THE NORMAL WATER LEVEL WAS ESTABLISHED IN THE REACTOR PRESSURE VESSEL, THE FEED WATER REGULATOR EQUIPMENT, WHICH REMAINED WITHOUT POWER, COULD NOT INTERRUPT THE WATER FLOW. THIS LED TO AN OVER-SUPPLY AND THEREBY TO AN INCREASE IN THE PRESSURE IN THE REACTOR PRESSURE VESSEL. THE PRESSURE INCREASE NOW CAUSED THE 14 SAFETY VALVES IN THE PRIMARY STEAM CIRCUIT TO OPEN AND TO LET OUT, FIRST, STEAM THEN PRIMARY WATER INTO THE CONTAINMENT. AT THIS POINT, THE CONTAINMENT WAS ALREADY, AUTOMATICALLY, TIGHTLY CLOSED BECAUSE OF THE SUDDEN PRESSURE DROP IN THE PRIMARY STEAM CIRCUIT. ACCORDING TO CALCULATIONS FROM THE DRAWINGS, ABOUT 200 CUBIC METERS OF WATER WERE RELEASED INTO THE CONTAINMENT THROUGH THE SAFETY VALVES. THE STEAM RELEASE PRODUCED A PRESSURE INCREASE IN THE CONTAINMENT OF ABOUT 0.30 BAR AND AN INCREASE IN THE TEMPERATURE

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ON THE AVERAGE OF ABOUT 60 DEGREES CELSIUS...AUTOMATIC BUILDING SPRAY CAME ON...CORE SPRAYS IN PRESSURE VESSEL CAME ON...IN ALL ABOUT 450 CUBIC METERS OF WATER WERE DUMPED INTO CONTAINMENT...TO A HEIGHT OF ABOUT 3 METERS....DURING INSPECTION OF PRESSURE VESSEL ON JANUARY 15 AND 16, 1977, ... DAMAGE TO ONE SAFETY VALVE WAS OBSERVED....SUMMARY!...ALL TECHNICAL SYSTEMS AND FACILITIES IMPORTANT TO SAFETY FUNCTIONED WITHOUT ERROR...CAUSES FOR THE INCIDENT WERE A DELAYED REACTION OF PART OF THE TURBINE CONTROL AND A BRIEF POWER FAILURE AT THE FEED WATER CONTROL STATION. THESE ARE COMPONENTS ALSO PRESENT IN CONVENTIONAL POWER PLANTS. END INFORMAL TRANSLATION OF EXCERPTS.... NO FURTHER DATA IN THIS REPORT ABOUT THE OPERATION OF VALVES.
STOESSEL

POOR ORIGINAL

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Department of State TELEGRAM

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PAGE 01 BONN 00951 181252Z
ACTION OES-06

INFO OCT-01 EUR-12 ISO-00 IO-13 ACDA-10 CIAE-00 INR-07
L-03 NSAE-00 NSC-05 EB-07 FEAE-00 DODE-00 PM-04 SS-15
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TO SECSTATE WASHDC 4783
INFO ERDA WASHDC
ERDA GERMANTOWN
USMISSION EC BRUSSELS
USMISSION IAEA VIENNA

UNCLAS BONN 00951

DEPT PASS TO NRC

E.O. 11652 N/A
TAGS: ENRG, GW
SUBJECT: NUCLEAR INCIDENT IN GERMAN GUNDREMMINGEN
NUCLEAR POWER STATION.

REF: A) BONN 7376; B) BONN 13986

1. SUMMARY. ON JANUARY 13, 1977 THE 237 MWE-REACTOR AT GUNDREMMINGEN HAD TO BE SHUT DOWN DUE TO THE RELEASE OF RADIOACTIVE STEAM INTO THE SAFETY CONTAINMENT OF THE REACTOR. OFFICIALS STATED THAT AT NO POINT HAD THE STAFF NOR THE POPULATION BEEN IN DANGER. IT IS HOPED THAT THE DAMAGE CAN BE REPAIRED WITHIN ONE OR TWO MONTHS. END SUMMARY

2. THE CAUSE FOR THE FAILURE WAS A BREAKDOWN OF THE 220 KV POWER LINE BETWEEN AUGSBURG AND ULM. A SHORT CIRCUIT RESULTED IN THE AUTOMATIC FAST SCRAM OF THIS BOILING WATER REACTOR. IN SUCH A

POOR ORIGINAL

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CASE THE POWER TURBINE IS TAKEN OUT OF THE CYCLE AND THE HIGH PRESSURE STEAM IS FED DIRECTLY INTO THE CONDENSOR. HOWEVER, BEFORE THIS PROCESS WAS EFFECTIVE, THE SAFETY VALVES OF THE PRIMARY CYCLE OPENED AND RELEASED LARGE AMOUNTS OF RADIO-ACTIVE STEAM INTO THE CONTAINMENT TO REDUCE THE STEAM PRESSURE. THE TEMPERATURE ROSE FROM NORMAL 90 DEGREES FAHRENHEIT TO 180 DEGREES FAHRENHEIT AND SUBSEQUENTLY MORE THAN TEN FEET OF RADIOACTIVE WATER WERE LEFT IN THE 100 FOOT DIAMETER VESSEL.

3. ON JANUARY 16 THE GUNDREMMINGEN DIRECTOR ANNOUNCED THAT THE TEMPERATURE HAD DROPPED TO ABOUT TEN DEGREES ABOVE NORMAL, THAT ALL WATER HAD BEEN PUMPED OUT AND DECONTAMINATED. HE CONTINUED THAT THE REPAIR WORK WAS TO START THE NEXT DAY AND WOULD INVOLVE THE DECONTAMINATION OF THE MECHANICAL AND ELECTRONIC EQUIPMENT. HE DENIED ANY SPECULATION THAT THE ACCIDENT HAD BEEN CAUSED BY HUMAN FAILURE AND SAID THAT AN INVESTIGATION WOULD DETERMINE WHETHER MALFUNCTION OF THE SYSTEM COULD BE PREVENTED BY MORE ADVANCED DEVICES.

4. EMBASSY COMMENT: THIS MOST RECENT INCIDENT IS ONLY ONE OF SOME THIRTY INCIDENTS IN THIS REACTOR, GERMANY'S FIRST COMMERCIAL REACTOR THAT BEGAN OPERATION IN 1965. MOST OF THESE INVOLVED CRACKS OR LEAKAGE IN THE COOLING SYSTEMS (REF B). THE MOST SEVERE INCIDENT LED TO THE SCALDING OF TWO WORKERS REPAIRING A STEAM VALVE (REF A). ALTHOUGH THE ACTING GUNDREMMINGEN DIRECTOR REFUSED TO SPECULATE THAT THE INCREASE IN REPAIR WORK AND SUBSEQUENT POOR AVAILABILITY WERE CAUSED BY A BAD DESIGN OF THIS FIRST GENERATION REACTOR, HIS PREDECESSOR RECENTLY COMPLAINED ABOUT THE DIFFICULTY IN MAINTAINING THIS REACTOR.

STOESSEL

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APPENDIX C

1. Memorandum for V. Stello from D. Eisenhut, undated, Subject: Summary of Conference Call.

2. Preliminary Evaluation - KRB 13 January 1977 Loss-of-Load

3. KRB Incident of January 13, 1977

POOR ORIGINAL

EXH. #6

Abnormal occurence at the nuclear power plant Gundremmingen (KRB), January 13, 1977

The boiling water reactor was supplied by General Electric/ Allgemeine Elektrizitäts-Gesellschaft, first criticality was achieved in 1966. The unit has a nominal power of 252 MWel, it is similar in design to GE-Plants Dresden - I/USA, Garigliano/Italy and Tarapur/India.

After an outage of the 220 kV grid on Jan. 13, 1977, the plant's turbogenerator generated the on-site power supply. A shortterm failure in the turbine control caused a speed decrease which was followed by an impulse for full opening of the turbine control valves. The fast increasing steam consumption initiated the isolation of the containment building and the scramming of the reactor (simulation of a steam pipe leak). Simultaneously, signals were initiated for start up the isolation condenser and for changing from on-site to off-site power supply via the 110 kV grid.

In order to compensate (increase) the reactor water level, the feed water supply was activated by hand and the feed water, control valve opened by hand. The power supply of the feed water control valve had failed due to the above mentioned changing of the power supply. Rendering the water control valve stuck in the open position.

To cope with such power failures, a pressure accumulator allowing full opening or closing is provided at the control mechanism of the feed water control valve. The energy stored in the pressure accumulator, however, was already used-up by the above mentioned opening of the valve.

Due to the opened feed water control valve, the normal reactor water level was relatively rapidly overfed, and the 14 primary safety valves (4 inches diameter) were automatically opened in consequence of the pressure increase (total amount of 200 t). The excess primary coolant discharged into the isolated containment building which then retained the slightly radioactive steam/ water mixture. All required safety systems as the emergency diesel generators, the containment spray system (which delivered 200 t), the core spray system (which delivered 40 t), and the air discharge from the containment annulus, performed adequately.

Illumination wiring, protective floor paints and a primary safety valve (ruptured between fitting and flange connection) were damaged by the escaping coolant.

The excess coolant accumulated (490 t) in the containment sump had an activity concentration of 40 mCi/Cubicmeter, the sump water was stored in the water treatment plant for radioactive decay. Presently it is in discharge in a controlled manner after adequate retention of the radioactive material.

POOR ORIGINAL

In venting the containment building on january 17, 1977, about 1 % of the licensed valves for airborne discharge rates (225 Ci/h) were discharged via the stack.

The heat removal from the reactor was assured at any time. The occurence had no consequences on the environment of the nuclear power plant.

REPORT TO BRAVARIAN PARLIAMENT, JANUARY 19, 1977

INFORMAL TRANSLATION

DRAFT

On January 1, 1977, about 2117 hours a significant incident occurred at the 237 EMW nuclear power plant of the RWE and the Bayernwerk AG (KRB I) in Gundremmingen during which weakly radioactive steam and weakly radioactive water ~~was~~ were released from the primary circuit into the containment.

According to investigations conducted to the present time by the Bavarian State Ministry for ~~Development~~ Regional Planning and Environment Problems and by the State Office for Environmental Protection, the course and the consequences of this significant incident occurred as follows:

On January 13, 1977, at 1834 hours a disturbance first occurred on the 220 KV high voltage line of the Lech Elektrizitaetswerke AG to Meitingen. The line was cut off by the switch ~~of~~ at the nuclear power plant. About 2117 hours a further disturbance occurred on the 220 KV high voltage line to Voehringen, which was similarly switched off by the line switch. The causes for these disturbances were ~~the~~ an extreme ~~danger~~

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drop in temperature and high air humidity which led to breakage of ~~many~~ central porcelain insulators. ~~Thus~~ the nuclear power plant was not damaged by the first disturbance ~~their~~ and its operation could be ~~continuously~~ continued; the consequence of the second disturbance was the complete separation of KRB I from the high voltage net. This type of separation of the nuclear power plant from the ~~net~~ grid by opening of the line switches and the ~~broken~~ breaker switches in the KRB I switch yard has occurred often during the 10 year period of operation of the KRB I. ~~At that time~~ Then the turbo generator of the power plant was always switched automatically to the production of the station power required for the nuclear power plant. In the present case, through a delayed reaction of the turbine regulator, a sudden drop in the pressure of the primary steam circuit was produced. The pressure drop caused the pressure sensor which monitors the primary steam circuit to react and to initiate independently the following protective measures: The reactor shut itself down quickly, all air ducts and pipes which penetrate the containment ~~xxxx~~ were shut tightly, so that no radioactive materials could get outside in the atmosphere or in the waste water in the case of a significant incident. The emergency condensor switched itself in to

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conduct away the after heat from the reactor pressure vessel. Because steam flow to the turbine was interrupted by the closing of the valve in the primary steam switch, ~~[illegible]~~ the station power supply was switched over to an independent 110 KV net of Lech Elektrizitaetswerke AG from the 220 KV high voltage line. To carry out this switching process, power consumers in the nuclear power station were automatically switched off within a few seconds and then immediately switched onto the new power supply. This brief interruption (2.8 seconds) ~~was~~ also caused the emergency power diesels to supply come on line, although they were not necessary in the present case.

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[illegible] of these remotely controlle[d] switching operati[on]s the feed water control station remained without voltage. The remote valves which control the feed water to the reactor pressure vessel were in the open position. Hereby the feed water pumps ~~could feed~~ which had been put in the circuit again could fe[ed w]ater in the reactor ~~vess~~ pressure vessel (about 18 cubic meter per minute). When the normal water level had been reached, the d[ ] water control station failed to interupt the water supply which resulted in an overfeed and a subsequent rise in pressure in the pressure vessel. This over-pressure caused the fourteen safety valves in the primary circui~~k~~t to open and let out steam first and then primary water into the safety containment. At this moment this safety containment had already been ~~locked~~ [illegible] due to the sudden drop in pressure in the primary circuit (steam). According to the construction plans the amount of water that was discharged through the safety valves into the safety containment was calculated to be about 200 cubic meter. The discharged steam raised the pressure by about 0.36 bar and the temperature to about 60 degrees celsius. Because of the rise in pressure in the safety containment the sprinkler automatically responded which is designed to condense the steam instantaneously and thus to break down the over pressure. During this incident a total of 450 cubic meters of water were released into the safety containment by the feed water pumps, the core sprinkler and the containment sprinkler. the lower part of the safety containment was hereby flooded ~~with/about~~ to more than three meters.

A more detailed examinination of the numbers will have to be carried out in the following investigations of the Bavarian Ministry for Regional Development and Environment and of the experts of the Technical Surveillance Agency Bavaria (TUEV-Bayern).

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[illegible]iological effects

As mentioned above the safety containment was hermetically locked, when the low-radioactive steam and water were discharged out of the primary circuit ~~into the~~ through the safety valves. Thus the released radioactivity could not get out into the environment. In the following davs the water was pumped out of the lower part of the saftey vessel into sEveral reservoirs of the waste watre purification system. Here the short-lived radionuclides that cannot be presipitated by chemical processing will remain to decay and the watre will be decontaminated.

The 22000 cubic meters of air contained in the safety vessel were analyzed and could be discharged out of the 110 meter chimney ~~following~~ [illegible] effluent guidelines. Such a discharge of radioactivity - [illegible] consisting ~~mainly of Xenon~~ to 93 percent of Xenon 133 - is in the order of magnitude of the ~~operational~~ ROUTINE discharge of the reactor and does not represent a hazard to the population. A rough estimate predicts a dose of fractions of millirem for a person who lives in the most unfavorable site near the reactor.

The discharge of the air ~~was~~ started Jan.17 at 11 a.m.. It was monitored ~~continuously~~ continuously with an additional discontinuous measurement every two hours to guarantee that the effluent GUIDELINES were met. The propagation was controlled by ~~a~~ parallel measurements of the meteorological ~~conditions~~ and the other conditions influencing radio immission. The results showed that the propagation characteristics wre optimal and that only a minor increase OVER ~~of~~ the natural level of radioactovity could be measured at ~~and~~ the ~~[illegible]~~ immission maximum.

The dose rate inside the reactor containment in the halls and the stairways rose from normal 1 millirem per hour to 5 to 10 millirem per hour. The staff could enter the reactor containment FOR SHORT PERIODS OF TIME without any imminent danger OF ~~of being exposed to~~ high direct radiation.

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Thirty minute REPAIR shifts could start on Jan. 15 at 7 p.m. with heavy protective gear and oxygen masks. The integral body dose of the involved personel was less than 10 millirem. Starting on Jan.18 the access to the safety containment is free and the repair and decontamination work is to start immediately.

Consequences of the incident and further measures.

First investigations of the consequences of the saftey vessel showed damages on the electricity installation, the paint on the floor and the walls, and the insulation of the steam pipes near the safety valves ~~plus~~ AND a damage on one of the valves. Since the safety containment is accessible without any additional protective measures, the installations will be cleaned dried and decontaminated. Subsequently all involved electrical and mechanical equipment has to be checked for damages.

Parallel to the tasks ~~the~~ involved authorities have started an investigation of the causes of the incident. The TUEV Bayern will monitor the numerous examinations for the re-start of the reactor and will prepare the final report on new safety systems that will have to be installed to exclude any such incident. The Gundremmingen reactor will only resume ~~the~~ operation after the final approval is given by the Bavarian Ministry for Regional Development as the licensing authority.

Before the investigations about ~~an~~ the Gundremmingen ~~accident~~ incident have been concluded, no new operating licenses will be issued to assure that any new findings about the reliability of the safety systems will be incorporated in the design of new reactors.

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Summary

During the incident at the 237-MW Gundremmingen reactor all saftey relevant system worked faultlessly. The safety containment was sealed off EVEN before the saftey valves responded so that no radioactivity was released. This was proved by immediate measurements of the invloved authorities. At no point of the incident the population was in danger. There were no personal injuries in the incident. The causes for the incident were a delayde response of one part of the turbine control and a short interruption in the voltage supply of the feed water control station. these types of control elements are also part of conventional power stations .
The reactor may not resume its operation unless the appropriate authority gives is approval.

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PAGE 01 BONN 00951 181252Z
ACTION OES-06

INFO OCT-01 EUR-12 ISO-00 IO-13 ACDA-10 CIAE-00 INR-07
L-03 NSAE-00 NSC-05 EB-07 FEAE-00 DODE-00 PM-04 SS-15
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R 181247Z JAN 77
FM AMEMBASSY BONN
TO SECSTATE WASHDC 4783
INFO ERDA WASHDC
ERDA GERMANTOWN
USMISSION EC BRUSSELS
USMISSION IAEA VIENNA

UNCLAS BONN 00951

DEPT PASS TO NRC

E.O. 11652 N/A
TAGS: ENRG, GW
SUBJECT: NUCLEAR INCIDENT IN GERMAN GUNDREMMINGEN
NUCLEAR POWER STATION.

REF: A) BONN 7376; B) BONN 13986

1. SUMMARY. ON JANUARY 13, 1977 THE 237 MWE-REACTOR AT GUNDREMMINGEN HAD TO BE SHUT DOWN DUE TO THE RELEASE OF RADIOACTIVE STEAM INTO THE SAFETY CONTAINMENT OF THE REACTOR. OFFICIALS STATED THAT AT NO POINT HAD THE STAFF NOR THE POPULATION BEEN IN DANGER. IT IS HOPED THAT THE DAMAGE CAN BE REPAIRED WITHIN ONE OR TWO MONTHS. END SUMMARY

2. THE CAUSE FOR THE FAILURE WAS A BREAKDOWN OF THE 220 KV POWER LINE BETWEEN AUGSBURG AND ULM. A SHORT CIRCUIT RESULTED IN THE AUTOMATIC FAST SCRAM OF THIS BOILING WATER REACTOR. IN SUCH A

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CASE THE POWER TURBINE IS TAKEN OUT OF THE CYCLE AND THE HIGH PRESSURE STEAM IS FED DIRECTLY INTO THE CONDENSOR. HOWEVER, BEFORE THIS PROCESS WAS EFFECTIVE, THE SAFETY VALVES OF THE PRIMARY CYCLE OPENED AND RELEASED LARGE AMOUNTS OF RADIO-ACTIVE STEAM INTO THE CONTAINMENT TO REDUCE THE STEAM PRESSURE. THE TEMPERATURE ROSE FROM NORMAL 90 DEGREES FAHRENHEIT TO 180 DEGREES FAHRENHEIT AND SUBSEQUENTLY MORE THAN TEN FEET OF RADIOACTIVE WATER WERE LEFT IN THE 100 FOOT DIAMETER VESSEL.

3. ON JANUARY 16 THE GUNDREMMINGEN DIREC OR ANNOUNCED THAT THE TEMPERATURE HAD DROPPED TO ABOUT TEN DEGREES ABOVE NORMAL, THAT ALL WATER HAD BEEN PUMPED OUT AND DECONTAMINATED. HE CONTINUED THAT THE REPAIR WORK WAS TO START THE NEXT DAY AND WOULD INVOLVE THE DECONTAMINATION OF THE MECHANICAL AND ELECTRONIC EQUIPMENT. HE DENIED ANY SPECULATION THAT THE ACCIDENT HAD BEEN CAUSED BY HUMAN FAILURE AND SAID THAT AN INVESTIGATION WOULD DETERMINE WHETHER MALFUNCTION OF THE SYSTEM COULD BE PREVENTED BY MORE ADVANCED DEVICES.

4. EMBASSY COMMENT: THIS MOST RECENT INCIDENT IS ONLY ONE OF SOME THIRTY INCIDENTS IN THIS REACTOR, GERMANY'S FIRST COMMERCIAL REACTOR THAT BEGAN OPERATION IN 1965. MOST OF THESE INVOLVED CRACKS OR LEAKAGE IN THE COOLING SYSTEMS (REF B). THE MOST SEVERE INCIDENT LED TO THE SCALDING OF TWO WORKERS REPAIRING A STEAM VALVE (REF A). ALTHOUGH THE ACTING GUNDREMMINGEN DIRECTOR REFUSED TO SPECULATE THAT THE INCREASE IN REPAIR WORK AND SUBSEQUENT POOR AVAILABILITY WERE CAUSED BY A BAD DESIGN OF THIS FIRST GENERATION REACTOR, HIS PREDECESSOR RECENTLY COMPLAINED ABOUT THE DIFFICULTY IN MAINTAINING THIS REACTOR.

STOESSEL

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KRB INCIDENT OF
JANUARY 13, 1977

SEQUENCE OF EVENTS

1. LOSS OF OFFSITE POWER (ICING/SHORTING)
   a. LOST MEITINGEN 220kv LINE [redacted]
   b. LOST VOHRINGEN 220kv LINE
2. ERROR IN TURBINE CONTROLLER AS STATION DROPS TO HOUSE LOADS AFTER INITIAL HIGH TURBINE SPEED
   a. LOW TURBINE SPEED/EXCESS BYPASS
   b. CONTROL VALVES OPEN - SIMULTANEOUS LOW STEAMLINE PRESSURE AND HIGH STEAM FLOW
   c. REACTOR TRIP/CONTAINMENT ISOLATION
3. LOSS OF AC DUE TO SYNC OF 110kv
4. FEEDWATER CONTROL SYSTEM NOT TRANSFERRED TO ALTERNATE OFFSITE POWER DUE TO 380 VOLT FUSE
5. D/G STARTED (NOT CONNECTED) - SAFETY SYSTEMS WORKED
6. FEEDWATER CONTROL VALVE FULLY OPENED
   - 1 SHOT ACCUMULATOR ON VALVE (NO POWER DUE TO FUSE)
7. ALL 14 SAFETY VALVES OPEN [redacted]
8. CONTAINMENT PRESSURE INCREASES (5 PSI FROM 200 CUBIC METERS OF WATER)
9. CONTAINMENT SPRAYS INITIATE (ADD ANOTHER 250 CUBIC METERS)
10. ECCS INITIATE
11. FEEDWATER PUMP MANUALLY TRIPPED

ATW NEWS – Fr: The Federal Republic of Germany – Feb. 1977

Grundnennungen

TOTAL ELECTRICITY GENERATION BY GERMAN NUCLEAR POWER STATIONS IN 1976: 24,751 GWH

The 12 German nuclear power stations in operation end of 1976 (with a total gross power of 6459 MWe) had a total gross electricity generation in the calendar year of 1976 of 24,751,114 MWh (gross generation in 1975: 21,858,782 MWh). The KNK facility with a power of 21 MWe was out of service throughout the whole year because of its conversion into a fast reactor. Three nuclear power stations with a total power of 2965 MWe were newly commissioned last year: GKN, KKB, Biblis B. The cumulated gross generation of all 13 plants by December 31, 1976 was 102,287 GWh. Here is a breakdown into the contributions of the individual plants (in MWh): Biblis A, 5,437,080; KKS, 5,461,288; KWW, 3,840,774; KWO, 2,335,820; GKN, 2,120,083; KWL, 1,703,090; KRB, 1,278,977; KKB, 1,085,534; Biblis B, 818,833; MZFR, 443,254; AVR, 119,514; VAK 106,867.

STATUS OF GERMAN NUCLEAR POWER STATIONS END OF 1976

In late 1976, 13 nuclear power stations with a total gross capacity of 6459 MWe were in operation in the Federal Republic of Germany, i.e., three plants with a total capacity of 2965 MWe more than at the end of the previous year (10 with 3494 MWe). The plants newly commissioned were GKN-1 (855 MW), KKB (806 MW), and Biblis B (1300 MW). By late 1976, 14 plants with a total capacity of 14,396 MW were under construction (or had been granted construction permits) (in the previous year it had been 12 plants with 11,975 MW). Four units of these newcomers with a total of 5346 MW were granted construction permits in the course of 1976: Grohnde (1361 MW), KRB II (two units of 1310 MW each), and Brokdorf (1365 MW). Another twelve units with a total of 15,213 MW (previous year: 14 units with 17,959 MW) are in the planning stage; this figure includes the new projects of KRL and KKI-2 with a total capacity of 2600 MW. This adds up to a total of 39 nuclear generating units to be commissioned by 1984/85 with an approximate total capacity of 36,110 MWe.

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ATW NEWS - Fr: The Federal Republic of Germany - Feb. 1977

KRB INCIDENT CAUSED BY GRID FAILURE

Following a series of defects caused by the weather in the 220 kV high voltage grid between Augsburg and Ulm on January 13, 1977 after 6.30 p.m. (rupture of insulators because of the extreme cold), an uncommonly abrupt failure of the grid at 9.17 p.m. caused an incident in the 252 MWe KRB Nuclear Power Station of Gundremmingen, in the course of which automatically opening safety valves allowed a radioactive water-steam mixture to flow into the containment. Lighting mains, decontamination paint, and a safety valve were damaged. Throughout the incident there was no hazard for the public in the environment of the plant. Also inside the nuclear power plant nobody was injured.

The Gesellschaft für Reaktorsicherheit (GRS) of Cologne has released the following additional information: After the power failure the turboset ran in the so-called isolated mode of operation as planned. A brief fault in the turbine control system resulted in a drop in the turbine speed, which was followed by a pulse causing the turbine governing valves to open all the way. The strong increase in steam extraction initiated an isolation valve of the containment (penetration valve with a venting seal) and a reactor scram. At the same time, connection of the auxiliary condenser and operation of the plant load supply system on the 110 kV grid were triggered. For correction (i.e., raising) of the reactor filling level a feed water line was connected and a water regulation valve opened. The power supply of the feed water regulation valve was faulty in the basement switching panel so that the residual energy stored in a pressure accumulator had been consumed by the previous opening action and the valve remained in the open position. Because of the wide open feed water regulation valve the normal filling level of the primary system was exceeded relatively quickly, and the safety valves were automatically opened as a result of a pressure increase. The excess coolant flowed into the fully closed containment which completely retained the slightly radioactive water- steam mixture. All the safety systems actuated as a preventive measure, such as emergency diesel power systems, building spray system, and annular suction system, worked correctly. The overflow collected in the sump of the containment had a specific activity of 4×10^{-2} Ci/m^3. The excess feedwater is presently stored in the water treatment system for decay and, following decontamination, will not be re-used but discharged under controlled conditions. In the process of air sweeping of the containment on January 17, 1977, less than 1% of the officially authorized levels for stack discharge were encountered. Cooling of the reactor was in no danger to fail at any point in time.

SNR-2 NOW WITH 1300 MW

The Europäische Schnellbrüter-Kernkraftwerksgesellschaft mbH (ESK) and the Internationale Natrium-Brutreaktor-Bau-Gesellschaft mbH (INB) in late 1976 agreed upon the power data on which the further development of the planned LMFBR demonstration power plant, SNR-2, will be based. According to the agreement reached, the net electric power, which had been 2000 MWe in the planning study drafted in 1975, will now be 1300 MWe. The continuous thermal net rated load is indicated as 3418 MWth, the linear rod power as 413 W/cm.

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Exh. #7

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ACTION NRC-07

ACTION

INFO OCT-01 EUR-12 ISO-00 OES-06 FEA-01 ACDA-10 CIAE-00
INR-07 IO-13 L-03 NSAE-00 NSC-05 EB-08 DODE-00
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UNCLAS SECTION 01 OF 02 BONN 01869

DEPT PASS TO DR. LAFLEUR, NRC, AS QUICKLY AS POSSIBLE

E.O. 11652: N/A
TAGS: ENRG, TECH, GW
SUBJECT: GUNDREMMINGEN NUCLEAR POWER PLANT INCIDENT

REF: A. BONN [illegible] B. LAFLEUR/MCCLELLAND TELCONS

1. EMBASSY POUCHING ASAP THREE REPORTS PROVIDED BY FRG MINISTRY OF INTERIOR (BMI):

A. REPORT TO BAVARIAN PARLIAMENT BY BAVARIAN MINISTRY OF REGIONAL PLANNING AND ENVIRONMENTAL PLANNING, JANUARY 19, 1977,

B. TELEX REPORT FROM BAVARIAN MINISTRY TO MINISTRY OF INTERIOR, AND

C. PRESS RELEASE: MUNICH, JANUARY 14, 1977.

2. EMBASSY WILL INCLUDE OUR INFORMAL TRANSLATION OF 1. A. REPORT.

3. NONE OF THESE REPORTS INCLUDE TECHNICAL DATA ABOUT REPORTED VALVE FAILURE.

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4. DR. WEIL, BMI, REVIEWED WITH SCIENCE COUNSELOR THE DATA HE PROVIDED TO FAULKNER, NRC, BY TELEPHONE ON JANUARY 31. THESE DATA FOLLOW:

ALL FOURTEEN SPRING-OPERATED PRIMARY CIRCUIT PRESSURE RELIEF VALVES OPERATED, AS INDICATED BY BURST DIAPHRAGMS . PIPE CONNECTING ONE VALVE TO CIRCUIT APPARENTLY RUPTURED AT -- OR NEAR -- WELDED SEAM CONNECTING VALVE TO PIPE. ABOUT THREE-FOURTHS OF PIPE CIRCUMFERENCE IS RUPTURED, AND VALVE NOW SITS AT AN ANGLE. VERTICAL PIPE NOZZLE AT THIS POINT IS NARROWING FROM 165 MM TO 94 MM, INSIDE DIAMETER. BMI BELIEVES SEAM -- OR ADJACENT PIPE -- FAILED BECAUSE NOT ONLY STEAM BUT EXCESS WATER WAS PRESENT IN REACTOR VESSEL DUE O OVER-FEEDING BY FEED WATER PUMPS THROUGH OPEN, REMOTELY-OPERATED VALVE WHICH COULD NOT BE CLOSED FOLLOWING 2.8 SECOND SHUT-DOWN FOR SWITCHING STATION POWER TO INDEPENDENT HIGH VOLTAGE LINE.

5. REACTOR OPERATOR HAS SUBMITTED REPORT ON INCIDENT TO BMI WHICH IS BEING STUDIED. FACTS, AS VERIFIED, WILL BE INCLUDED IN OFFICIAL BMI REPORT WHICH WILL BE FORWARDED TO NRC THROUGH EMBASSY ASAP AFTER APPROVAL BY MINISTER, PROBABLY IN FEW DAYS.

6. INVESTIGATION CURRENTLY DELAYED BY DECONTAMINATION OPERATIONS USING STEAM. ENTRY INTO CONTAINMENT WILL AGAIN BE POSSIBLE IN FEW DAYS AND INVESTIGATION WILL BE COMPLETED THEN.

7. BMI PLACES NO LIMITATION ON USE BY NRC OF DATA OUTLINED IN PARA 4, ABOVE. THESE FACTS WILL BE PUBLISHED SOON.

8. FIRST OF THREE REPORTS (SEE PARA 1. A., ABOVE)

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WAS PASSED TO EMBASSY "NOT FOR DISSEMINATION WITHOUT APPROVAL OF THE BMT OR THE NRC." HOWEVER, BMI SUBSEQUENTLY WITHDREW THIS LIMITATION - AS MARKED ON COPY POUCHED.

9. FOLLOWING ARE EXCERPTS OF REPORT MENTIONED IN PARA 1. A., ABOVE, WHICH RELATE TO OPERATION OF PRESSURE RELIEF VALVES DURING INCIDENT. NUCLEAR POWER PLANT WAS CUT OFF FROM GRID, THEN:

BEGIN INFORMAL TRANSLATION OF EXCERPTS: QUOTE ... DELAYED REACTION OF TURBINE REGULATOR RESULTED IN SUDDEN PRESSURE DROP IN PRIMARY STEAM CIRCUIT ... PRESURE SENSOR REACTED...SHUT VALVE IN PRIMARY STEAM CIRCUIT...SWITCHED STATION POWER FROM 220 KV LINE TO

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