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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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SUBCOMMITTEE ON CLASS 9

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NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON CLASS 9

ACCIDENTS

Tuesday, April 26, 1983
1717 H Street, N.W.
Washington, D.C.

The Subcommittee on Class 9 Accidents met at 8:30 a.m., pursuant to notice, William Kerr, the Subcommittee chairman, presiding.

PRESENT FOR THE ACRS:

W. KERR

C. SIESS

P. DAVIS, Consultant

J. LEE, Consultant

M. CORRADINI, Consultant

A. WANG, Designated Federal Employee

G. QUITTSCHREIBER, Designated Federal Employee

AUDIENCE PARTICIPANTS:

R. BERNERO

O. BASSETT

C. KELBER

M. FLIESHMAN

P R O C E E D I N G S

MR. KERR: The meeting will come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards, specifically the Subcommittee on Class 9 Accidents.

I'm William Kerr, Subcommittee Chairman. Other Committee members present today are Mr. Siess and Mr. Shewmon. Our consultants present are Messrs. Lee, Davis and Corradini.

We are here to continue our review of the severe accident research program, and today we will concentrate on continuing performance, status of source term work, recent activities and decisions in a severe fuel damage program.

The meeting is being conducted in accordance with provisions of the Federal Advisory Committee Act and the Government in the Sunshine Act. Harry Quittschreiber is the designated federal employee. Alan Wang is also present.

Rules for participation in the meeting were announced as part of the notice of the meeting published in the Federal Register of Friday, April 8, 1983.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register notice.

I request that each speaker identify himself or herself, and use a microphone.

We have received no written statements from members of the public, nor have we received requests for time to make oral statements.

We will proceed with the meeting. In context, what we are doing is a continuing review and an effort to put this research program into context in which -- or understand the context which fits into the Nuclear Regulatory Commission's ongoing program of dealing with severe accidents.

In my own review of the most recent version of NUREG-0900, the Nuclear Power Plant Severe Accident Research Plan, I'm reminded that on page 1-2, the third paragraph, there is a statement that if meaningful requirements for severe accidents are to be developed, a rational structure for decision-making is needed. Safety goals and the numerical guidelines pertaining to NUREG-0880, offer a useful criteria to judge whether modifications of existing requirements -- whether modification of existing requirements is necessary. Probabilistic risk assessment establishes the formal, logical methodology to be used in evaluating severe accident safety issues in terms of the safety goal. The limitations and us- -ness of PRA in the context of severe accident analys- -st be carefully established if this methodology is to be used correctly in the revision or confirmation of current regulatory requirements.

I interpret this to mean that the foundation for axiomatic, or however one puts it, basis for this approach is an effort to meet safety goal requirements, and from a set of criteria which depends on probabilistic risk assessment. I am willing to have that paragraph interpreted differently. If my own interpretation is incorrect, I would hope to have some light shed on the subject. And I think also it is perhaps useful in the discussion of individual elements of the program to comment on how they do or do not fit into this sort of context.

The agenda that I have indicates that the status of containment performance will be the first item to be discussed, and I have Arlotto, Bernero, and Vollmer, in that order. But I don't know what order the speakers will come. Bernero indicates it will not be alphabetical, and he will lead off.

MR. BERNERO: Dr. Kerr, the first thing I would like to do is speak a little bit about that citation you just made from NUREG-0900, the use of PRA in the decision process.

The question is frequently raised about the use of a safety goal, the use of PRA, the establishment of numerical criteria, and using them for decision-making, and there is a great temptation to do this in what is apparently a rigorous way. That is, setting numerical criteria, and then slavishly running a PRA and taking the results and saying it meets, it doesn't meet, making some pro forma uncertainty analysis to persuade yourself how well you meet or how badly you don't meet the criteria.

We don't think that the state of the art for PRA is at that stage, and I don't think the Commission does, either, because the Commission has put out its safety goal for trial evaluation, and I think the best way we can describe it is as using PRA and attendant safety goal calculations as a logical structure for laying out the relationships, the thought process, the interactions between the factors associated with some safety issue or some problem, and then superimposing on that logical display discrete use of judgment. A very good example is before us today. Some of the people in this room are involved with the ATWS question. You, yourself, are the chairman of the ATWS subcommittee. As you know, you yourself can approach that problem, and we have for some ten or more years -- you can approach it probabilistically and chase your tail for years, trying to calculate an exact number for the probability of the event or failure to respond to the event, failure to mitigate the consequences of the event, and yet we do not have a good, solid criteria against which we would consider one is acceptable, one probability is acceptable or another is not.

What we must do is look at the level of probability of trip -- or failure, transient without trip, occurring; look beyond that to the likelihood of the plant getting new, unacceptable plant conditions. And as you know from the discussion there, even the definition of "unacceptable plant conditions" is extremely hard to do.

In a pressurized water reactor, is service level C a go/no-go threshhold. We look at that. We can't say that it is absolutely acceptable to be below service level C pressure and absolutely unacceptable to be above it. What we really say is at service level C pressure, we suspect we have lost confidence in our ability to rely on valves reopening for HPI injection; we have lost our confidence that large numbers of steam generator tubes won't have ruptured. And to some modest extent, we may even have lost our confidence in the structural integrity of the whole reactor coolant system.

Others argue that you could make those same points at service level D, which is a goodly number of psi beyond service level C.

So, lacking acceptable criteria, a very well-defined criterion, it would be foolhardy of us to attach overly great significance to the exactly calculated pressure or the exactly calculated probability of exceeding that pressure.

In the severe accident regime, whether it be ATWS or something else, we are trying as carefully as we can to develop a logically displaced relationship and estimate of all of the factors that go with severe accident consideration and then use them, use this displayed relationship in our decision process, being very careful not to hang our hats on that bottom-line number. That is a real hazard. It is a real hazard in this.

And the more you slush around in PRA and safety goals, the more vulnerable you are to talking yourself into acceptance of something or rejection of something solely on the basis of how it meets a calculated number.

MR. KERR: Mr. Bernero, I was not trying to defend or attack this paragraph. But in my efforts to understand the thrust of the severe accident research program, I need some context of how one knows what to do about severe accidents, and as I read the description of the severe accident research program, a significant amount of effort is being attached to refining PRA, so that presumably it will be more useful. And when I read this pragraph, it seems to say -- whether it represents the view of the Staff or not, I don't know -- that the judgment as to whether the pla- s acceptable will be based to a considerable extent or fety goals and that the way one determines whether a plant meets safety goals is by use of PRA.

Now, it may not say that, and if it doesn't say that, then I need some -- I would welcome, I should say, some additional elaboration on where it is we are, where it is that we are headed, and how we will know when we get there.

As I say, this is what I read in 0900, and if that is not the true gospel --

MR. BERNERO: It is true to the extent --

MR. KERR: Then I need an exegesis.

MR. BERNERO: It is true to the extent that, as I think I've tried to tell you before, there is a systematic effort in the severe accident research program to look at all the risk information available to us on all the plant types, and to the extent possible identify classes of reactors, their accident sequence characteristics, and those features or alterations of design that would improve either by preventing the dominant accident sequences or by mitigating their consequences.

And, yes, this is PRA. In fact, this is the most difficult of PRA that WASH-1400 merely asserted it could do; that is, taking a single plant, the PRA of a single plant, and rising to generic conclusicns drawn from that that affect a class of plant.

Looking at Grant Gulf and saying from that I can derive judgments about the risk of BRR-6's with Mark III containments, solely on the basis of evaluating Grant Gulf.

Sometimes that is not an unreasonable extension of the calculation; sometimes it is an unreasonable extension. The attempts we are making to refine PRA -- a little later on when I get to talking about containment failure -- we are looking at the PRA information available to find out what are the generic challenges to different types of containments that have to be assessed. We are trying to be very careful not to say that we are calculating a single number, a single matrix of dominant accident sequences quantitatively and laying that out as the sole basis of judgment.

MR. KERR: Are you telling me, really, that is what this paragraph means, what you have just said?

MR. BERNER: Yes. That is what the program is doing. That paragraph is attempting to express the fact that constructing this analytic quantitative analysis framework and for illumination of the issue comparing it to the safety goal calculations which the Commission has out for trial use now, to have that on the table before you, let you exercise judgment in so much better a way than merely not having that and looking at the issues with the murkiness of qualitative logic.

MR. KERR: Who finally is going to exercise the judgment?

MR. BERNERO: The Commission.

MR. KERR: Please, please.

MR. BERNERO: Ultimately but pratically, the collegial staff.

MR. KERR: Somebody at some point has to decide that a class of reactors needs to be changed or not.

MR. BERNERO: That a certain feature might be warranted in a class of reactors, and it is really a collegial exercise of the technical staff that would then be presented to the Commission for ratification, which is what you normally do in rule-making. You don't wait around for the Commission to tell you exactly what the technical content is.

MR. KERR: So at this point you don't know how you will decide, but you will get people together in a room, given the results of the research program, and given the PRAs, and then it will be sort of a committee of the whole decision as to whether something does or does not need to be done. But in the meantime, some people who have the reactors won't have any way of knowing whether their reactors are okay or not.

MR. BERNERO: Well, they have in a sense a parallel analysis of the same issues using very similar methodology through the IDCOR program.

MR. KERR: The decision, as you tell me, is not going to be based on the analyses and any set of criteria. The decision is finally going to be made by the judgment of this group of NRC people presently undefined?

MR. BERNERO: After reflection on the analytical result, yes, and I would assume the industry does the same thing.

MR. KERR: But their judgment won't be the judgment of the group.

MR. BERNERO: Certainly not. Not necessarily, I should say.

To go to another example, it is going on right now in White Plains, New York, the hearing on Indian Point, is an ample display of the very same kind of thing. There is probably now no plant in the world that has enjoyed as much quantitative risk assessment as Indian Point. Indian Point Units 2 and 3 have been deeply analyzed by the owners through specialized contractors. That analysis has been subjected to very, very close scrutiny by the Staff and its contractors, and there has been a complex interplay of quantitative risk analysis and judgment that has led the owners in some cases, and Harold Denton in other cases, exercising his responsibility as Director of Reactor Regulation, to say go and fix that control room roof and go and put some sort of fire barrier here, and go do something else, whatever it is, selecting those changes which constitute justifiable alterations of the Indian Point facilities in order to permit further operation of them.

Now, those are not exactly calculated solutions. They do not exactly calculate what does it take to lower core melt frequency to less than 10 to the minus 4.

MR. KERR: Are you telling me that the Indian Point proceedings will form a model for the decision process to be used on operating reactors?

MR. BERNERO: To a very great extent, yes. Indian Point is the severe accident decision process in microcosm, because it happens to be the most popular site licensed for operation in the U.S, because it happens to have two relatively large reactors already licensed at that site, it is running ahead of the severe accident decision.

MR. KERR: Would you guess, or could you guess with some reasonable confidence when a decision is likely to be reached on Indian Point?

MR. BERNERO: Well, it is difficult to say how long it will take the hearing process to complete, and then, of course, the Commission will undoubtedly not speak on it until the hearing process is complete, and I would set aside the emergency response issues right now.

MR. KERR: ~~e reason I asked is that it would appear to me that that ~~cision might well be reached without benefit of much of the severe accident research program. Is that not the case?

MR. BERNERO: Yes and no. That decision -- if I take that decision at the threshold of Harold Denton's rulings, which I think is a reasonable first threshold, if you take the director of NRR's rulings, that decision is current. It is an early 1983 decision, and it has the benefit of some severe accident research. It also has the benefit, as I said earlier, of the largest plant specific risk analysis that has ever been done. So he has the benefit of that information, and to a certain extent you can say a regulatory severe accident decision has been made in Indian Point with respect to the plant design. There is the off-site emergency response issue, as I say. I have to set that aside.

But the information between now, say, and this time next year is absent in that. The Commission will have the benefit of that. I've just -- I just roughly would assume that by the time the hearing is concluded and the Commission has a chance to look at the case and draw their conclusion, that it might be as much as another year from now. But again, Indian Point is a pretty good example of the severe accident decision process, just laced throughout with probabilistic risk analysis, laced throughout with the explicit consideration of safety goal levels, safety goal criteria.

MR. KERR: It is also laced throughout with a lot of other things that neither one of us will mention. But I was trying to learn something about the relationship of this research program to the decision-making process, and I don't learn much about that from the Indian Point case.

MR. BERNERO: I can only bring you back to this.

MR. KERR: Assuming that most of the severe accident research is in the future, which I have to assume.

MR. BERNERO: Yes. A good deal of the physical phenomena research still lies in the future, as the word we often don't like to use, confirmatory research. If you would make a decision this year, early this year, end of next year, there are still physical phenomena research that you won't have the benefit of and you will be making judicious estimates just as we did in ECCS. I hope we are not as conservative as we were in ECCS. Maybe we will be. I just don't know.

But again, if you just look at the Indian Point, you can't afford to look at every plant out there as specifically and as exhaustively as we are looking at Indian Point.

MR. KERR: As I say, what I'm trying to do is to understand how the research program fits into the decision-making process, and I can't gain much of an insight by looking at Indian Point. I will simply have to wait and see how things develop to understand. Because at this point, I don't. But why don't you go ahead with your comments.

MR. BERNERO: All right.

Let me talk today in this continuation of severe accident research. I will be speaking to you first about the assessment of containment failure. I have some Vu-Graphs on that. There are representatives of the appropriate sections, branches, divisions of NRR and of the Office of Research here, as well, to answer questions. Then, after that, I want to talk to you about the source term program office and what the Staff is doing within the severe accident environment or situation that is labeled with source term in order to bring it to a head a little more effectively, a little more quickly.

Let me go first to the assessment of containment failure.

MR. SIESS: I hate to start in so early, but I'm not going to listen to an assessment of containment failure until somebody defines "failure" for me. There are too many different definitions going around. I want to know what you a re talking about.

MR. BERNERO: Containment, of course, is that outer barrier other than the reactor coolant system, that is supposed to contain the fission products in the event of mishap.

As you undoubtedly know, the risk significant failure of that containment need not be catastrophic bursting; it can be large leaking.

As I go on, I will be talking about two models for containment failure: a threshold model which would characterize containment failure as on or off. You either have containment or you blew it, you know, that you reach some pressure, some limit, at which point the containment just bursts open and is in catastrophic failure.

An alternative, which is probably much more realistic, is a leakage model, whereas the pressure increases the characteristic of the containment is to have a higher and higher leak rate, perhaps accellerating into catastrophic failure, or perhaps not. Perhaps coming to some degree of blowdown that blunts the pressure increase that prevents catastrophic failure.

Now, there is one chart that I regret I didn't use. I should have brought it along. Jim Meyer made a chart for his Indian Point analyses that is very, very illustrative. It shows the off-site consequences for different containment failures. That is, catastrophic failure, total rupture of the containment, eight-inch pipe equivalent failure. Just assume that there was an eight-inch pipe somewhere that blew open, that that is the model he used; a four-inch pipe somehwere, assume that blew open to the atmosphere, and a half-inch pipe. And he calculates the off-site consequences. And what you see from that display -- it was a very nice matrix -- you see that you don't have to have catastrophic containment failure in order to have significant off-site risk. An eight-inch pipe will certainly compete with the catastrophic containment failure using current models, and a four-inch pipe failure can even get you there. It turns out that a four-inch pipe failure is also approximately at the level that will prevent further pressure buildup. A four-inch pipe on most of the pressure transients will give you this kind of a controlled blowdown.

MR. SIESS: Am I correct that four inches is about 100 percent a day leakage?

MR. BERNERO: I would say several.

MR. SIESS: Half-inch is probably somewhat greater than the 1 percent a day?

MR. BERNERO: Somewhere between a tenth and one.

MR. SIESS: That is a good answer.

The point I think we have to appreciate is that the containment, although it is a structure, it is not a structure like a building; it is a structure like a tank. The containment is a tank and when it leaks, it fails.

MR. BERNEF Yes. If it leaks badly enough, it fails, and it is a graded failure, and that makes the problem a little more difficult.

MR. SIESS: If it leaks at all, it fails. It's just degrees of failure. Again, you can define "failure" at various levels.

MR. BERNERO: I am not interested in an exact threshold of containment failure. I am interested in the distribution of risk from that containment.

MR. SIESS: I hope you are not interested in exact pressure, because you ain't going to get it.

MR. BERNERO: The important thing is that I must understand the containment characteristics. Let me postulate two types of containment. One containment is very unlikely to over pressurize, but when it does, it fails catastrophically. Another containment is much more likely to leak at the range, say, of 10 to 100 percent per day; it is more likely to leak at that, but has great resilience, will not overpressurize and fail catastrophically.

MR. SIESS: What you are calling fail catastrophically could be better expressed as having an infinite leak rate. If you plotted pressure versus leak rate, at any pressure there is some stoichastic distribution of leak rates. At zero pressure it is probably zero. At 1 psi, it is not zero. Right?

MR. BERNERO: Yes. That's right.

MR. SIESS: At something like three times the design pressure, you know, it is very likely to be infinite, to several hundred percent a day leak rate. It is a distribution and it varies with pressure, or ratio of pressure to design pressure.

MR. BERNERO: I could draw a curve here. If I had leakage -- and let me just make this like a logarithmic scale in percent. This is percent per day. I could have 1, 10, 100 and 1,000, and if I take pressure --

MR. SIESS: Do you want to use the sketch I've got in front of me?

Do you want to use mine?

MR. BERNERO: Yes. It may be the same thing.

If I take pressure up to something like two-and-a-half times design pressure, I can draw something like an asymptote there. I say leakage at zero pressure is zero and have some sort of characteristic that goes like that.

MR. SIESS: On each one of those verticle slices you could draw your distribution?

MR. BERNERO: Yes. There is some uncertainty band that goes with that, and a little later on I'll talk about it.

I wish I knew that curve. That curve would be a real joy to have. Unfortunately, that is not the way people treat containments.

MR. SIESS: That's the way they think about them.

MR. BERNERO: That is an essential way to think about themk if you can get there.

MR. SHEWMON: Why is pressure versus leakage -- probability distribution would be a problem, but pressure versus leakage must be fairly straightforward, isn't it?

MR. BERNERO: No, it isn't. It is not a fixed orifice.

MR. SIESS: The probability of a large hole is greater the higher the pressure.

MR. SHEWMON: Okay.

MR. BERNERO: Even the shape of that curve --

MR. KERR: It seems to me the way to fix that is to put in a hole so you know what is going on.

MR. BERNERO: Let me make a couple of points before we go on.

The importance to risk to containment integrity is very great. The best known phenomena for the attenuation of fission products, settling, and all the attendant plateout mechanisms are found in the containment. There is a greater level of confidence in what aerosols will do, what the radionuclides will do to play out or settle out in the containment atmosphere than there is in the reactor cool[illegible] can go back to WASH-1400; you can go to any risk assessment done since then, and if you look at the nature of the problem, you can easily understand that.

The very high temperatures, the very severe physical conditions inside the reactor coolant system, make it very difficult to predict, or to conduct experiments to show what happens to radionuclides in that environment.

On the contrary, when you get into the containment atmosphere, where you are at manageable temperatures, a few hundred degrees Fahrenheit, pressures of just a few atmospheres, there is abundant evidence as to what aerosols do. So our models for fission product attenuation are much better and our confidence in those models is much higher for containment mechanisms.

Now, if containment fails early, or, as is the case I'll show you later, in some reactors, where the containment might fail before the core melts, then you have a different problem. You have in effect removed that nice attenuator, the containment from the equation more or less, and you must rely for attenuation of fission products solely on mechanisms that work in the reactor coolant system, and it makes it a lot more difficult to predict with confidence, and therefore it is generally true that early containment failure will dominate the risk when you look at containment response to core melt accident sequences.

Now, there are two aspects of this problem.

You have to look at the certainty or uncertainty one has of --

MR. KERR: Excuse me. Let me try to understand that statement, if I can.

Early failure will dominate risk sort of means that you have made some assumptions about both probability and consequences.

MR. BERNERO: Indeed, yes.

MR. KERR: In WASH-14400 did early failure dominate risk?

MR. BERNERO: Yes.

MR. KERR: In what sense?

MR. BERNERO: Let me take a simple example --

MR. KERR: I'm not trying to disagree with you.

MR. BERNERO: A large, dry containment, with some sort of core melt sequence, say, a station blackout. There are two challenges to containment that you must face in that large, dry containment. One is, where the core starts to melt, it boils off some steam first, and the containment goes up to some initial accident pressure, and then there is a melt-through and you get a steam spike. There is a release of energy from the reactor coolant system to the containment atmosphere, generating some gases, but principally steam in that case, and you get a pressure pulse. And that press re pulse, if it is strong enough or high enough to either lead to catastrophic failure of the containment or substantial leakage. If it does, the radionuclides at that time are at a high enough concentration as to have severe off-site consequences.

If, on the other hand you get through that first hump --

MR. KERR: In a PWR sequence, what sequence are you talking about?

MR. BERNERO: Station blackout, as an example. If you don't fail in that pulse, you know, the melt-through spike, if you get through that, you look at the heat transfer equations and you find that the pressure will dip as you condense on the walls of the containment and things like that and hit the sinks, and then with no containment cooling, you will ultimately build up to a failure, oh, eight hours, twelve hours, fifteen hours later, after the accident started.

If that failure occurs, at that time you have had all those hours of containment, settling mechanisms to work, and the consequences are vastly lower.

MR. KERR: I understand that. What I was trying to get at, do you include the interface of LOCA in the early containment category?

MR. BERNERO: Yes, although it is also a moot point. Interfacing LOCA bypasses containment and has to be dealt with in a different arena altogether.

What I'm saying here --

MR. KERR: When you talk about early failure, you are not talking about the interfacing LOCA?

MR. BERNERO: No. I'm talking about mechanisms where the containment has a role, and it is the early failure of containment that dominates the sequence.

MR. KERR: This comes back to my next question which is, what do you mean by "containment failure"? You don't mean interfacing LOCA --

MR. BERNERO: No. It is not even in this arena. You work on interfacing LOCA by depressing --

MR. KERR: You are talking about core melts.

MR. BERNERO: Core melt sequences in the containment where the containment has a role. I'm not talking about interfacing LOCA. The important thing is that the difference in consequences between early failure of containment and late failure of containment is so great that even a modest fraction of sequences going to early failure can dominate risk, because there can be orders of magnitude difference.

Now, as I was saying, looking at containment failure, the challenge to containment and response to containment have to be considered.

Usually people use the word "containment loading" or "loadings."

MR. KERR: Excuse me. I'm trying to educate myself, so you have to forgive me.

Is that statement very strongly dependent on what might develop out of the source term research? Is it possible that one might find the source term for early failure to be different than what we now think, whereas the source term for something else may not be much different, so that one might modify this?

MR. BERNERO: Yes, indeed, you might.

Let me give you a dramatically different postulation. A boiling water reactor, because of its very nature, has to have steam dryers and separators in the reactor vessels. So it's got the world's supply of metal over the core's head, so to speak. Now, in that configuration, let me postulate for the moment that a careful scrutiny of core melt and fission product transport scenarios would show that under virtually every circumstance, a boiling water reactor core will melt and all he goop will come out and go into that acres and acres of metal that is above it and stick and that the attenuation of the reactor coolant system would therefore be very high. Then early failure of containment becomes almost academic. The early or late failure of containment will have relatively little effect. The source term work, if it demonstrated such a phenomena to be true with some confidence, would have shown that there was relatively little burden laid on the containment to mitigate the consequences of the accident. But that has to come true. That is a postulation.

MR. CORRADINI: Could I ask a question?

Again, it is more the introduction that I'm still trying to understand. If I follow that logic, then the next thing I worry about, since everything tries to go to disorder in such an event, wouldn't one worry about all that structure now that one has captured all those fission products, melting?

MR. BERNERO: Yes. I would have to carry that postulation on and say that the melting phenomenon going through the bottom head would not release or melt down that structure. The stuff would be left there, safely captured in the steam dryers. It wouldn't come true unless that worked.

MR. CORRADINI: And your example would not carry through to the PWR simply because of the amount of surface area is so drastically different?

MR. BERNERO: Some argue that the PWR, with all that forest, the control rod tubes, has a sufficient area to do the same thing. The results we have seen so far suggest that we aren't seeing overwhelming plateout inside the reactor coolant system, at least on some accident sequences.

MR. CORRADINI: I ask just one other thing. When you say "early failure dominates risk," you gave one example of a PWR station blackout, and you said steam spike, or some rapid rate of rise of presusre due to steaming in the accident.

What is another means? The only other thing that comes to my mind is the hydrogen.

MR. BERNERO: Now, on the station blackout, it is argued you can't burn the hydrogen because of steam inerting and no source of sparks, but you can have LOCA sequences that could have hydrogen generation that would lead to igniteable or combustible hydrogen in the containment. It could give you a spike.

The important thing is that any mechanism that will fail the containment before you have had substantial benefit of the plateout mechanisms that work in the containment, and this is going to be true unless further investigation can show that the plateout mechanisms or attenuation of fission product mechanisms in the containment are not as important as we now think they are. If we can show that primary system, that is, reactor coolant system attenuation is very significant, that in turn will diminish the signific nce of attenuation in the containment, but until we can show that we have to recognize that the containment is the predominant mechanisms for attenuation of fission products.

Now, in order to appraise the containment performance --

MR. KERR: You have just convinced me. I'm not going to remove the containments.

MR. BERNERO: I'll be the last to recommend their removal.

The two aspects of the problem that you have to look at are the loads imposed on the containment, and this is how to describe the generation of steam, hydrogen gas, other noncondensible gases that releases that, together combine to challenge the pressure volume capability of the containment.

MR. SIESS: Including temperature?

MR. BERNERO: Yes. Ultimately a containment is a tank; it is a tank-like vessel, and the energy that it holds is going to be translated into tank challenge, namely, a pressure volume. It has a certain volume.

MR. SIESS: I'm thinking of temperature per se.

MR. BERNERO: Yes. The temperature per se, example, the study of the Brown's Ferry containment where it was discovered that certain electrical penetration assemblies would become thermoplastic. They are currently designed for the LOCA design pressure and temperature, but at a substantially higher temperature they will ooze and extrude out of the penetration hole and give multiple leak paths.

MR. SHEWMON: I have had the suspicion for some time that though it is more fun and faster and easier to do experiments at high temperature, that safety lay in dumping water on this thing, whether it was inside the pressure vessel, under the pressure vessel, spread around the containment or whatever. I never seem to hear that discussed in the United States. It is just station blackout and apparently the workers are sent home and we all sit there and watch, or at least that seems to be the scenario. I was heartened that things might be at least better in France the other day when the group said that their procedures were to look at how they could get water into the containment if that was -- sorry -- into the pressure vessel if that was still there, to get water into the sump if that was where it was, to get water in to flood the containment, if that is where it was. And what they could do to get auxiliary power if indeed there was station blackout, and how far they would have to bring that.

Does that sort of thinking ever penetrate over here? I haven't heard of it if it has. I would be greatly heartened if you could say yes.

MR. BERNERO: Yes, it does. That sort of thinking is on the table in this whole process of severe accident.

Charlie Kelber will be here to talk about this specifically a little later this morning. He will talk right after my talk or a little bit later. I don't see him here yet.

MR. SHEWMON: When you talk about auxiliary station blackout, do you ever talk about blackout for five hours or ten hours, or is it just gone forever?

MR. BERNERO: Let me invite your attention to a pace-setting, or a precedent-setting ASLAB ruling here in the United States, that has underneath it the whole tone of this.

St. Lucie case, all of us think of Florida as one long extension cord with all of the people at the very end. It very nearly is that. It is not quite as unstable as we think.

In any case, down on Hutchinson Island, the St. Lucie plant, in its licensing process, got into the issue of the reliability of off-site power and the potential for blackout, and there was a good deal of liscussion of what is the probability for blackout, how reliable are the AC and DC power sources. The plant is a PWR with the traditional turbine-driven auxiliary feed pump. How long can you run on the DC, you know, without AC power? And in a nutshell, the resolution of it was that the appeal board ruled that station blackout is a design basis event.

Now, you have to be very careful. That doesn't mean design basis accident, class 1 through 8, show that no single failure, you know, that whole ritual that we do in chapter 15 of the final safety analysis report. No. What they meant, and what it really means is it is a design basis event; that is, it is something that you should consider in the design, and that plant and every other plant now also, I believe, has included in its emergency response procedures considerations of, I know you are not supposed to lose AC power, but you did; what are you doing; what are the extraordinary options open to you to restore AC power, to provide continued cooling of the plant, to maintain the plant so that you don't melt the core. And this ranges all over the map from jury-rigging off-site power to jury-rigging cooling.

You know, normally in a PWR, you are going to have a turbine-driven pump that runs on exhaust steam drawing from some condensate supply. Then you start opening the door to fire pumps and things like that.

I am reminded that years ago I worked in the Naval Reactor Program on surface ships. We had a hose connection on top of the reactor compartment, with a portable fire pump. When all else failed, you did that. That is in the extremest emergency procedures.

MR. SHEWMON: Do we have dedicated standpipes in any U.S. plants?

MR. BERNERO: I can't answer that. I don't know.

I do know that they have fire systems -- you know, like these pumpers, you know, usually a fire truck only holds about 500 gallons a minute. But most plants have some source of water supply nearby, like a pond, a river, or something like that. But they do have these procedures.

Now, I can't tell you how many plants actually have gone into the extent of deliberately flooding the containment, having an emergency procedure that would deliberately flood the containment.

MR. SHEWMON: It should be relatively easy at Indian Point.

MR. BERNERO: It should be relatively easy at most any of them.

MR. SHEWMON: They've done it once.

MR. BERNERO: I just want to make the point that you've got to look at both the challenge to the containment and its response to that challenge in order to decide whether or not you have failure.

Now, if you look at loading, containment loading, what we are doing, what I'm describing to you now is an approach by which we are trying to assess containment failure in the broadest sense for purposes of estimating the risk of the plant or class of plants. The first thing we have to do is take this plant, which is a surrogate for a class -- Grand Gulf is the surrogate for the BWRs or whatever, identify its dominant accident sequence characteristics. What are the challenges, what are the chains of events that are challenging this containment; then calculate the containment loadings and then you run into a problem. When you calculate the containment loadings, the most dramatic, and I'm sure you have heard it many times, you get into physical models of core melt where we don't really know what happens, and remember what the MARCH code suffers for all of its vices.

When the core melts, we are talking about a room-sized block, actually greater room-sized block, a hundred metric tons, uranium oxide, zirc alloy and all the other fittings. You've got over a hundred metric tons of material, trying to form a molten ball, glob, or series of globs through the bottom of the reactor vessel, after getting through a puddle of water in there, and then falling through a sump that may be dry or wet, depending on the accident scenario.

The steam spike, that aspect which raises the question of early containment failure, is going to depend on how intimately that molten material mixes with the water. What is going to happen is, globs, maybe as big as a basketball, maybe as big as a baseball, molten material are going to fall into the water. There will be a fracturing of that material that will enable smaller and smaller pieces to mix with the water, transfer their energy into the water, making steam, and that steam will be the cause of the steam spike.

Now, these codes, the MARCH code, do not calculate that fracturing. You tell the code from physical phenomena knowledge, from what you know about how core M behaves when it falls into the water, you tell the code, assume that whole mass of a hundred metric tons breaks into particles of average diameter, one inch, or half-inch, or as recently -- we have run some sensitivity analyses, .04 inches. I honestly don't even believe that one. But somewhere in the range of several inches down to fractions of an inch. And depending on the particle size you choose for that fracturing, or distribution of size, you will get a different steam pressure. You will get a different energy transfer.

So you have to conduct sensitivity analyses for key uncertainties, and that is probably the largest single one.

MR. CORRADINI: I ask a question there, since this interests me.

Can't one automatically, just from thermodynamics, get an upper bound, if you just quench the core, if you know the maximum amount?

MR. BERNERO: Yes, you can. When you do a sensitivity analysis for particle size and take it down to little, fine pellets, you know, .04 inches or something like that, you are in a sense bounding it by saying I'll give it infinite heat transfer, I'll get all the energy out as a steam spike and see what happens, and maybe you can come out smelling like a rose. If you've got a sufficiently robust containment, then even total energy transfer might get you through.

MR. CORRADINI: I'm just trying to link it up with what I've read. That is the conclusion of the Zion PRA.

MR. BERNERO: The Zion PRA concluded -- well, there is more to it than that, as I'll show later. If you assume that the threshold of failure is a high one, and you are highly confident in that threshold. There is no big uncertainty tail on that. Then a simple thermodynamic bounding may get you by. You don't get early containment failure. You get late containment failure. But you've got to do this sensitivity analysis, and this is where it gets sticky. If you are going to convolute your uncertainty, if you are going to deal with your uncertainty in containment loading, and your uncertainty in containment response, you have to have some sort of probability distribution at least assigned to this range of loadings.

Now, if the thermodynamics are such that you can bound it, you can assign the probability 1 to the bound, less than or equal to that as a probability 1, and you might be home free. But this may not be the case for all containments.

For containment response, it is a tricky thing. We have a committee that the severe accident research review group has appointed -- this is the one that is chaired by Denny Ross, and it's got all the division directors affected, in NRR, research, and NIE -- and this committee is drawn from the cognizant people in both research and NRR in the areas of structural engineering, equipment qualification, and risk analysis. And it affects or brings up issues of the structural prediction of containment, the leakage of containment. Like the containment systems branch in NRR is deeply involved in this. The leakage expectation. Equipment qualification people for the expectations and the experience of electrical penetration response to overtemperature, to overpressure. This group has twofold objectives:to develop two models that we can use in source term calculations. Now, we have them listed here as the leakage-before-failure model and the threshold model, the difference being in an ideal world, the two are combined. If you have perfect knowledge, you actually combine the two with a single curve like that. But right now most risk analyses are done with a threshold model. If you look at Zion, you look at Indian Point, you look at many of the others, you will find a threshold model used which in effect says, here is a single pressure, and below that pressure the containment is sound. There is essentially no significant leakage, and above that pressure the containment has catastrophic leakage, and then one calculates containment loadings, and they are either below that pressure or above that pressure. If they are above, you call it failure. If they are below, it's success.

In actuality, we need to expand that model to consider substantial and perhaps extremely undesirable leakage. Going back to what I was saying earlier about six-inch or eight-inch pipes: Leakage that is not in the category of total collapse or burst to the containment but high enough to give off-site risk, that is very nearly the same. We are trying to develop both models and as well this committee is to recommend a plan of action for improved confidence in the models.

We have a number of programs going on right now that were put together based on the wisdom of a year, two years ago to analyze and to test the structural integrity of containment, the integrity of various boundaries in containment, penetration, seals, things like that, for equipment qualification.

These programs may be judiciously redirected to get additional information, some feedback from this work that would point out, for instance, a body of simple test information that might be readily available that can greatly change our knowledge of a model.

MR. SIESS: Bob, what do you know about the leakage that exists before the accident even starts?

MR. BERNERO: Well, this is part of it. The containment systems branch, in particular, and NRR, has been looking at the history of Appendix J testing, and what sorts of things are leaking, what sort of characteristics they seem to display. Looking at the LERs, for instance.

MR. SIESS: What are they learning? I saw a report the other day of a 66-inch purge valve that they couldn't even get the pressure on it. So obviously they don't know how much it is leaking. I've seen a lot of excessive leakage tests where that was the case. I mean, I wouldn't know where to put my distribution.

MR. BERNERO: When you get the substantial overpressure above the design, there are severe pressures. Some of them are double butterflies and so forth.

Is there someone from containment systems branch that would like to speak on that?

MR. SIESS: That's why I wouldn't start that curve at zero.

MR. BERNERO: Well, zero pressure, zero leakage. That's where I started.

MR. SIESS: It might be convection.

MR. BERNERO: For government work, zero is good enough at that level.

But this is a real problem. We do have some experience that leads us to suspect substantial leakage from some of the valves and particularly for large valves, the purge valves.

MR. SIESS: It seems to me you have to make a distinction also between the path outside the containment and the path to the environment.

MR. BERNERO: Yes. I will.

Would you use the microphone.

MR. HUANG: John Huang from Containment Systems Branch.

We currently have a request for proposal. We have sent it to four different labs. We have already received a proposal back from the lab and we are in the process of reviewing it.

MR. KERR: Excuse me. This is a request for proposal to do what?

MR. HUANG: To evaluate the reliability of the containment isolation system, because we had a feeling that the containment may not work the way they are supposed to. In other words, the containment may fail as a result of many things: seal failure, sometimes the area indicates containment leaks. Of course, it is difficult to estimate exactly how much it leaks. But we try to at least get some idea from this to see how reliable the containment isolation system is and what kind of estimate we can get from those leakage data. We have to make a lot of guesses. We hope we can get at least some idea from those data.

MR. KERR: This is not part of the research program? This is technical system program?

MR. HUANG: Yes. The containment system is issued in the hopes that we can get some idea how reliable the system is.

From the result of that program we hope we can find a means or ways of increasing the reliability through the testing program, or better design or some other way.

MR. SIESS: Let me suggest something: At least one operating plant operates with a small pressure in the containment at all times, which I think would detect a leakage in a purge valve, for example.

MR. HUANG: Yes, indeed.

MR. SIESS: I hope when they look at reliability they might see what effect that operating procedure has on reliability.

MR. HUANG: That is one of the things we have currently. We hope through these data, especially, like a leaking purge valve, we hope we can institute a leak-testing program. But we are waiting for it to see how reliable the containment design or system operation is before we impose that testing as a part of our current leak-testing program.

MR. KERR: Thank you.

MR. BERNERO: Well, for the containment response, as I say, we are trying to develop both models. The ideal would be a well-defined leakage-up-to-failure curve with the distribution at any given pressure, and then one could much more rigorously analyze risk for different scenarios. But that is going to be quite difficult to get. Here again, just as I said in containment loadings, you will find that there will frequently be a need to do a sensitivity analysis, and then through the use of expert opinion, convert those sensitivities into quantitative uncertainties, and deal with this mixture of uncertainties from containment loading and containment response as quantitative uncertainties.

Now, some of the early work we are doing -- I've got some rather long detailed Vu-Graphs I'm going to go into here.

MR. CORRADINI: I ask a question, out of curiosity.

Is there currently an accepted method as to choosing the threshold? Is it chosen such that one always is using a hundred percent per day, 200, 300?

MR. BERNERO: No. The threshold of risk significance is going to be found somewhere in the realm of several hundred, 200, 300 percent per day leakage. Somewhere in that range you are going to get significant off-site risk if you leak it at that rate.

The threshold model is a threshold of catastrophic failure, and the structural people -- and by this I mean reinforcing bar, steel, concrete, those people -- will, if that is all the question is -- somebody else will have to guarantee thermoplastic penetrations and all that, and seams, but if you just look at the physical structure of containment, there is a much better feeling about the threshold being somewhere from 2.2 to 2.5 times design pressure.

Now, I'll call that the macroscopic containment of containment. That does not guarantee that the electrical penetration assemblies or that the equipment hatch gaskets, or that the purge valve internals, the big butterfly valves, that those things won't separately fail at some pressure other than that. That is strictly speaking of steel shields, concrete, and rebar.

MR. KERR: Just a minute. I want to get another data point.

Do you agree with 2.2 to 2.5?

MR. SIESS: Sure. If you stick to first yield and last yield and not talk about large strains, and talk about the membrane, the things that he mentioned, it is not unreasonable. There is still aband on the confidence level.

MR. BERNERO: Yes. There is an uncertainty band on it. How well was the rebar placed and all of that? But the important thing that you should recognize is in risk assessments to date that has been the failure model used, and I have strong recollections back when we reviewed the Zion risk analysis right after it came out, discussing it with Dr. Siess in this room, the uncertainty curve which is put on it, which I'll call it a structural engineering uncertainty curve, and a very narrow, or tight one at that.

MR. SIESS: The yield strength of the rebar was the only variable on that curve.

MR. BERNERO: Yes. It was too sanguine, too optimistic. There was a big uncertainty. I have just ruled out the other things, like containment penetrations.

But basically on these slides I want to give you an example of what is being done. In the severe accident program element 5.1, we have the accident sequence evaluation program. It is looking at all the PRAs for dominant accident sequences, and you will find the pervasive typo, and especially in my division, with words like likelihood, I'm embarrassed with the lack of an E.

But the ASEP with its current results, we are looking at them to say, in this case, a Mark I containment for a boiling water reactor, that is roughly two dozen reactors with operating licenses, and looking at what are the dominanat accident sequence types, and in a nutshell the boiling water accident sequences boil down to two dominant types: TW and TC, which are generic terms for a transient sequence wherein loss of decay heat removal or containment heat removal is suffered, or a transient sequence with ATWS. And the two characteristics, or accident sequences, are somewhat the same. In the TW sequence what you have is a successful trip and cooling but a failure to cool the containment. So that the decay heat energy of the reactor is slowly but surely heating up the suppression pool to the point that the containment will reach an upper limit pressure of some sort.

So the energy rate in this case is decay heat.

MR. KERR: In the TC sequence, it is also assumed that the SL, SCLC, whatever it is, doesn't work.

MR. BERNERO: There are a variety of sequences.

MR. KERR: If you get to 30 percent --

MR. BERNERO: This is for TC. I'm just talking about TW now. I've had a trip. This is decay heat. Decay heat is warming up the pool.

In the ATWS sequence, you can generally assume that the recirculating pumps trip, which will cut your power from 100 percent to about 30 percent, but then the sequence has the characteristic that you didn't scram by pushing buttons, or you didn't borate it.

MR. KERR: It is more than not scramming. It is also a failure of the secondary shutdown system, as well.

MR. BERNERO: Failure to borate the system to shut down. So that now the only difference between this sequence and this one is that in the TC sequence, a faster, or more rapid energy transfer is taking place. Instead of decay heat, you've got power heat. You've got about 30 percent power. So that you have a more rapid pressure buildup.

But now in both cases, what you have is a sequence which on its face is going to fail the containment first and then melt the core. You see, what it is doing -- you haven't melted the core; you've got water in there, and you are pumping water in there, but the containment is rising up toa high enougn pressure to fail, and then upon failing can lead to disruption of piping or cavitation of vital pumps; it can lead to core melt, and then the core melt would occur after the containment has failed.

Let me show you on the next slide a rough sketch of the containment pressure load.

Now, sequence A, where the containment pressure just goes steadily upward and reaches some catastrophic failure point, is the one where the rate of energy input is such that you ar rising and there is no mechanism to turn it around. The core is not melted yet. And you just reach some failure pressure and away they go. You burst the containment. That leads to core melt failure. And very crude numbers for the containment failure point.

In an ATWS type sequence, it would be in about several hours. In a TW type sequence it is more like a day, because it is decay heat. That, of course, is plant-specific. It will vary with the exact size and pressure capability of the containment.

Now, there was an analysis done on the Brown's Ferry plant and published as one of the SASA analyses that followed sequence B, where the containment pressure would build up, and temperature as well -- it is a saturated containment -- and you would reach a point where sufficient leakage would be incurred by failure of penetrations, that you would turn over the pressure pulse, and the pressure pulse would be something like sequence B here, where you never quite get to catastrophic failure. You sort of blowdown through 20 or 30 pentration assemblies that have failed.

Now, if that sequence takes place, you've got to ask yourself a number of questions: Was nature kind to me? Did nature give me a blowdown that was sufficiently gentle, or was this perhaps abrupt? If this is an abrupt blowdown, remember the propensity for boiling water reactors to incur dynamic loads. You could have a loading situation where flashing of the pool could cause dynamic loads that could fail containment. You have to understand reasonably well how the blowdown occurs, the rapidity or abruptness of it, in order to have confidence if you were to model this as a safe relief. Noncatastrophic failure of containment, you would have sufficient confidence in your knowledge of the blowdown characteristic such that you wouldn't upset the general geometry of the suppression pool and cause catastrophic containment failure.

The last sequence is kind of drawn with a gap in it because you don't know what is going to happen. If during containment heatup degradation of pump performance leads to core melt before containment failure, you might have some third sequence that goes up and maybe even has a spike that goes up to give you catastrophic failure and them comes down.

Now, this responsive containment has to be understood and some fractional distribution, some probability distribution for the different failure paths or mechanisms is needed. This one up here, sequence A, is likely to dominate the risk because it fails the containment and then fails the core and gives you a core melt with essentially no containment, or badly disrupted containment around it.

This squence B might give you great mitigation of consequences if the blowdown is into auxiliary spaces which have significant fission product attenuation capability. It is almost like going into secondary containment. You have fire protection space available out there and things like that.

MR. KERR: Is it the consideration of the A that has led to the discussion of venting the Mark III, I guess it is?

MR. BERNERO: Or any of the Marks, yes. Really a filtered vent containment system is a controlled sequence B. It's sole purpose would be -- rather than suffer A, or C, which might be a variation of A, it says let me get a safe relief path that I can trust that will give me some substantial attenuation and I won't bring it in until I get up to some substantial pressure. I'll pick a pressure that is high enough that I won't casually use it and low enough that I won't burst the containment or blow out penetrations or go into some uncontrolled failure mode. A filter vent is just that.

MR. LEE: Could you perhaps tell me at this stage what kind of information you may need to be able to determine whether sequence A is more likely to occur or sequence B is more likely to occur, or when you think such information might be available?

MR. BERNERO: The principal information we need for sequence B relates to penetrations and seals, and there is very little available right now. We are looking very hard at that right now, and that is one of the areas where we might be able to get some crucially important information in the very near term.

Right now risk assessments in general and my own personal conviction is it is sequence A until proven otherwise. If I would put numbers on it, I would assign 90 percent probability to A and divide 10 percent probability between B and C.

MR. LEE: Even with so many penetrations that you have to consider typically in containment?

MR. BERNERO: Yes. That is what risk assessments have done. WASH 1400 just said it's A, and that's what we've done since then. Because not enough is known -- see, remember, if you have -- let me take for a moment the Brown's Ferry analysis where sequence B gave a substantial mitigation, where the blowdown through the electrical penetrations into the auxiliary buildings substantially mitigated the consequences.

I have now an event tree, and the event tree has -- instead of a yes, no, containment does fail, containment doesn't fail, I have two no's. Everybody knows the containment is going to fail. We are just arguing about which of the yes's applies.

If I have any substantial fraction of sequence A, it is going to dominate risk, again because it is early failure and it is failure directly outside essentially with no containment.

I have to have a very large proportion of the sequences end up with profile B in order to substantially reduce the risk.

MR. SIESS: Bob, if you've got A there, obviously uncertainty inthe level of that dashed line, or uncertainty in the slope of that solid line, doesn't really give you much of a problem, does it?

MR. BERNERO: No, because the core hasn't melted yet.

MR. SIESS: It is going to go fairly early and it's going to go?

MR. BERNERO: Yes.

MR. SIESS: Unless you are talking about venting at some point, you don't have to know where point A is very precisely?

MR. BERNERO: Only in this regard. In its TC sequence, ATWS, you have to do things in a hurry. On the TW sequence they go so long that the fire trucks and emergency procedures have plenty of time to work and it's good to know the tolerance of the system, that you've got 24 hours, or 30 hours, or something to work with. And the probability of recovery, even by Rube Goldberg methods, is very real.

So, insofar as you want to know that, it is important, you know, the options for recovery. But on the TC it happens pretty quickly.

In general, like in the ATWS consideration, we consider that the operator better do what he has to do in the first 20 minutes, 15 minutes. It is a rather demanding sequence of events, simply because there is so much more energy.

In the ATWS sequence you've got roughly 30 percent power dumping into that containment.

MR. DAVIS: Question: On sequence A, don't you still get the action of the suppression pool scrubbing of the fission products?

MR. BERNERO: It depends onwhere it busts. There is a question about that. Sequence A is going to take you to some high pressure which will give you what might be called a violent failure of containment.

Now, if you can expect that the suppression pool will be intact and have essentially all of its water still there, you might -- and the failure is in the top of the wet well somewhere -- then you might indeed get scrubbing afterward. Rather than count on that happening -- in a Mark III containment I would be more inclined to agree with you, that, you know, even if you burst the containment, naturally it might come out okay. On a Mark I containment I wouldn't have any confidence at all in it unless you went and built a rupture disk pipe, a vent pipe, on the wet well, the vapor space of the wet well, and put a rupture disk, or a pop-open valve or something like that to cut in at some less than catastrophic pressure in a way that wouldn't be catastrophic in itself. You wouldn't get some great boiling of the suppression pool to make it jump off the ground.

MR. SIESS: When you are talking about Mark III, what do you call the containment?

MR. BERNERO: The steel building. 1.2 million cubic foot, whatever it is.

MR. SIESS: The secondary?

MR. BERNERO: The wet well.

MR. DAVIS: The more recent figures I've seen on suppression pool scrubbing negativeness even under saturated conditions, if they are correct, series A may not be the dominant --

MR. BERNERO: If you can be sure it's there. In other words, that's what I'm saying: In order to be assured of that, you might very well have to provide a chosen failure path and failure scenario. In other words --

MR. DAVIS: For Mark I?

MR. BERNERO: Yes. For Mark I, what we are looking at in the severe accident research program is that very thing: to install on a Mark I that amounts to a simple duct, a pipe, with an opening mechanism that would open at a significant pressure but a pressure below, I'll call it catastrophic failure pressures, so that you would have a controlled pathway, and you would be left with what amounts to a filtered vent containment system wherein the pool itself is the filter.

MR. SIESS: Bob, on a Mark I, would the drywell and the wet well see the same pressure?

MR. BERNERO: With the vacuum breaker, yes.

MR. SIESS: But nobody has looked to see which one would go first?

MR. BERNERO: You don't know.

MR. SIESS: I know we don't know, but we are spending several million dollars down at Sandia to find out when a drywell will go.

MR. BERNERO: There are two pressure vessels. Take the Mark I. The drywell and the torus are two steel pressure vessels, and their design pressure is essentially the same. Their failure pressure is so congruent that you are swamped with doubt. That is why we think the only way -- see, this is the scenario you would want through a safe exhaustive path; namely the pool, and the only way we think you could be assured of that is to put the pipe where you want it --

MR. SIESS: I see what you are doing, but it is interesting, you see, rather than trying to compute the burst pressure for those, you say, well, we can put a system in that, will guarantee it will go through the second one.

MR. BERNERO: Yes, because that is the only acceptable one.

MR. SIESS: It's an interesting approach. I'm thinking about how I could apply that to a dry containment. I'm working on it.

MR. BERNERO: Of course, what we are looking at with the boiling water reactor is the vent alone, using the pool as the filter, and then the vent with the filter, and I might add that in the boiling water reactor it appears to be fare more attractive. The initial results indicate that the pipe alone, using the pool, because of that pool scrubbing data, the pipe alone appears to be attractive far more than the pipe with filter, you know, something like a sand and gravel filter outside.

MR. KERR: I'm sorry. You say the pipe with no filter is more attractive than the pipe with a filter?

MR. BERNERO: Yes. From a cost benefit standpoint, in the sense that the pool provides --

MR. KERR: You are talking about on a cost benefit basis?

MR. BERNERO: Yes. A cost benefit basis.

Because the pipe is so much cheaper. A pipe vent alone is on the order of a million dollars. A pipe with filter, or a filtered vent, 10 to 20 million dollars. It really goes up in cost. And you are working on the tail of risk.

MR. KERR: You are also giving thought to mechanism for deciding when that vent will vent?

MR. BERNERO: Yes. Not too low and not too high.

MR. KERR: I don't believe in the rupture disk.

MR. BERNERO: No. No one believes in the rupture disk, if for no other reason the pressure controlability of it, and the fact that when it goes it tends to be an abrupt failure, and you get a -- you know, the whole pool is going to start bouncing.

The one I just took off, I covered in the words.

MR. KERR: Out of curiosity, is the thinking at present, if you can discuss it, that the NRC would make that decision, that the governor would make a decision, or the utility?

MR. BERNERO: My feeling is you better build that decision into the plant. That is not a decision -- that is my personal feeling. That is not a decision that should be debated during that pressure rise. You know, the plant should have a control system of some kind, a simple, elegant, complex, whatever.

MR. KERR: The decision will be made automatically based on some parameters built into the plant, and neither the operator nor the NRC have the burden of sucking on a finger while the pressure builds up and youhave to decide whether to pull the plug on the containment.

The other Vu-Graph was an explanation of what I have already done.

If you turn to the PWR, this is a large dry. The important point to make here is large dry containments cover quite a spectrum.

Everyone uses the term "large dry" as if it is a standard. I'm almost trying to adopt not so large and not so dry or something, because some are much bigger than others. They range in pressure capability over quite a spectrum, and therefore you have to look carefully at the specifics of sets, at least, if not individual plants.

If you go into these systems, you will find a number of things from the accident sequence evaluation program and from current work.

One of them is this: Remember what I said about the importance of containment is greatest when the attenuation of the reactor coolant system is least.

Now, in a pressurized water reactor, there are some accident sequences which give you some reasonably good confidence that the reactor coolant system is going to get some of that stuff before the containment sees it. For instance, some of the loss of coolant accident sequences which are downstream of the steam generators -- just visualize, if alluviam from the core comes off, it has to go through the upper plenum, down through the hot legs. There is the enormous surface area of the steam generator in the way. So that you have some sequences wherein you have the potential for very significant reactor coolant system attenuation. However, you will have other sequences, the station blackout for example, TMLP prime, which in general -- remember, you have lost AC power. The heat buildup boils the water out of the reactor coolant system, into the reactor building, and what you will find from this sequence -- let me just switch to the next Vu-Graph because it's got a picture of it -- you will find the containment loading following a profile something like this. The steaming, as you come up, you pressurize the reactor building, depending on its size, to some level, and then you have the steam spike. The melt-through occurred somewhere in here, and you will have a steam spike due to the melt falling into the sump and converting at least some of its energy to steam. And, of course, the question is: Is this spike high enough to hit this line or not? and then you hit the heat sinks a little bit and you have a pressure drop, and still lacking heat removal from containment, you build up until you have failure of the containment structure.

Now, the risk is going to be dominated by whether or not that spike hits the containment failure, either catastrophic failure or substantial leakage failure. The same problem goes here. Will the purge valves collapse, fall open; will the equipment hatch -- remember how a PWR equipment hatch is. It faces the outside, and you've got about 100 linear feet of doorway with some kind of a gasket or seal there, and if you blew it all out, that is a big hole, you know, a very long slot.

So, that is the issue that dominates. And what you have to do is look at the containment loading, and your certainty of it, the containment pressure capability and your certainty of it.

Now, I pulled one Vu-Graph and put it aside, because I think it best illustrates the concept we are trying to pull in here. Right now we are doing some preliminary work wherein we are taking containment loading -- admittedly I think with a conservative bias to it, particle sizes. We are doing sensitivity analyses where particle sizes for the hot drop into the water range from an inch down to .04 inches, and with that distribution, assigning either a normal or a log normal distribution to the containment load pressure, taking containment failure pressure and looking at some sort of distribution around it that we think is a ittle more realistic than the ones we've seen, you can get an overlap of the tails, even in a large dry containment, and that is quite significant, because if you have an overlap of those two tails, it says even though your normal or central estimate of load pressure is below containment failure pressure, there is a finite probability that the mixture of uncertainties can combine and give you early failure.

Now, depending on the validity of the distributions you have, it is not too difficult for us to generate numbers that would approach 10 percent or more here for the combined probability of loading being high enough and capability low enough due to failures of one sort or another, to give you an early containment failure of risk significance.

This is why it is quite important for us to look at both the threshold model and the leakage-before-failure model in the containment failure.

MR. CORRADINI: Could I ask a question?

I was looking at your graph and just something struck me. Yesterday we had people from IDCOR here, and I asked what is early containment failure to them. They said minutes versus hours.

I look at your plot here,and here you have four to five hours where the Delta P may cause failure. Unless I'm misinterpreting, you are saying early.

MR. BERNERO: No. They were undoubtedly speaking of minutes after melt-through. Your time zero is usually time zero when the accident sequence started. It takes several hours to boil away the water and melt the core. The time for the heat to melt and go through the bottom of the reactor vessel. I'm quite sure when they said minutes, they meant minutes after that process, within, say, the first 30 minutes after the core melt drops to the floor.

MR. CORRADINI: I thought that. Maybe I should ask the question this way: then one doesn't only worry about the Delta P in that Vu-Graph you showed, but also about the rise time?

MR. BERNERO: Yes.

Now, from the standpoint of risk, the rise time is much more important for heat transfer considerations than it is for decay energy. You know, the decay heat and the nuclide concentrations are dropping, but they are not dropping that rapidly on this scale. But for heat transfer purposes, it is important.

MR. CORRADINI: I guess I have to go back -- I'm still trying to understand.

Then all the Delta Ps for the steam spike are of the order of minutes, all the rise times, from that Vu-Graph you showed?

MR. BERNERO: Yes. This is not a very good drawing.

MR. CORRADINI: I understand that. I'm just curious.

MR. BERNERO: This time is the time it takes to heat up the water, boil enough water away to uncover the core, melt the core, and attack the bottom head.

MR. CORRADINI: And that is four to five hours?

MR. BERNERO: An accurate construction of this curve would have a rise time in a very steep -- what, 20 minutes, 10 minutes -- well, you know, many minutes. Less than 30 minutes. Less than 30 minutes. Possibly even far less than 30 minutes, for the Delta P to occur.

In fact, that is why it is usually called a steam spike. On a more accurate plot it is much more vertical.

MR. CORRADINI: To follow through so I get complete understanding of this: The reasons that the minutes versus the hours is so crucial is due to the aerosol and fission products settling in the containment. So the time scale of that physical process is of the order of hours.

MR. BERNERO: Yes. The containment never sees the fission products until here, in bulk.

MR. CORRADINI: They only see what gets out through the PORV?

MR. BERNERO: They see what gets out through the PORV, but then when the melt-through of the head occurs, that is when the bulk of the radioactivity enters containment, and you get substantial aerosol generation and so forth, and then from the containment's perspective, they have a few minutes before the steam spike, and many hours for the late containment failure.

For the mechanisms in containment you need the hours here to do anything significant.

MR. CORRADINI: This is a little bit off the track, but I'm trying to understand the differences. So one could look at it two different ways: One is the Delta P and the rise time, and another one would be the uncertain -- the other one would be the uncertainty of how I produce the aerosols and the rate of aerosol production, given some Delta P and some rise time? It is not a well-known thing as to the rate of aerosol production and how it is formed once it leaves the vessel?

MR. BERNERO: If you go into the mechanics, you have an aerosol term generated from the melting of the core. Call it an aerosol flow trying to get out. When the core bulk gets out into the containment and starts to react with concrete and water, it starts generating another aerosol term that is superimposed and can actually scrub the original aerosol term. It can cause a new wave of agglomeration that can pull out aerosols that were put in there originally.

MR. DAVIS: Bob, there is a school of thought which postulates that these pressure spikes will be very unlikely because the reactor vessel lower head failure process will be dominated by a failure of tubes that go to the bottom of the reactor vessel, and you will get small streams of molten material dispersed widely throughout the containment.

Has the NRC Staff adopted any position about the likelihood of that scenario?

MR. BERNERO: No, not yet. Of course, that is true on most of the Westinghouse -- CE has the round bottom. It's Westinghouse and B&W have the pins, or flux monitor tubes.

Yes. That changes the rate at which the molten material goes out and might be a dispersal mechanism that would subdue the spike by changing the rate of transfer. That is, the rate at which the heat goes out. It could be that it gives you greater energy conversion. You know, it might accentuate the possibility of small particles and better heat transfer, albeit at a slower rate.

Now, we don't have a position now -- Jim Meyer, in NUREG 0850, Indian Point, you know the analysis of that containment; he ended up with the luxury of abcund. It is a big enough containment that he was able to get by with abound. He didn't have to sweat it. He wasn't that close. But we don't have a position yet, you know, that we would say the best judgment would describe the scenario in one way or the other.

MR. KERR: Bob, I'm looking for a good place for a ten-minute break.

MR. BERNERO: I was going to suggest about here. I'm about wound up with this.

What I would like to do is go into a discussion of the source term approach, and it is a much more general thing than this. I think it would be ideal to take it here.

MR. KERR: We will reconvene at 20 after.

(Recess.)

MR. KERR: What happened to my leading man?

We are ready when you are.

MR. BERNERO: Sorry to hold you up.

MR. LEE: Mr. Chairman, if I may, I would like to raise a question or two for Mr. Bernero.

MR. KERR: Before you raise a question, an earlier question had been raised about whether we need Containment Branch people here for any further purpose.

Do you need Containment Branch people here for any further questions or comments?

MR. SIESS: I think not.

MR. KERR: Thank you.

MR. BERNERO: Thank you, gentlemen.

MR. KERR: Mr. Lee wanted to raise a question.

MR. LEE: In one of your earlier Vu-Graphs which showed the probability densities for containment load pressure versus failure pressure, you said your overlap region has a probability of about 10 percent or so.

MR. BERNERO: Could. Yes.

MR. LEE: Could. Yes.

I'm just curious whether you think that 10 percent potential leak probability is too large or too small, or if you performed the proposed containment research and the further data-collecting process and so on, you can reduce the uncertainties associated with that 10 percent probability? Do you think that uncertainty could go down by an order of magnitude, or what do you think we should try to accomplish?

MR. BERNERO: For instance, we are talking about a pressurized water reactor with a large dry containment. If the probability is out where you don't have overlap of tails, the releases -- the delayed containment releases, slow overpressure, the release categories are like WASH 1400, PWR 6 or 7 categories. They are very mild consequences by comparison.

If you are in this overlap regime, you have a release category like PWR 2 or 3. And the result is the consequences rise dramatically. PRW 2 has early fatalities in it. PWR 6 or 7 do not.

The difference in consequences is so great that even a 10 percent chance of getting a PWR 2 release is a relatively significant and a dominant, in fact, fraction of the risk. But now you are talking about the level of risk that you have and is it tolerable. So that we are talking about a given reactor, say, or set of reactors where our best estimate of the risk is at a certain level, and take the example of Indian Point: You have an estimate of the risk at that level, and there is at that level an estimated probability of PWR 2 type releases, and you look at that and you have to make a judgment. That is the dominant risk, and is that at a tolerable level, and if not, what do I do? What are the alternatives to reduce that risk?

So there is no single probability that brings it into the threshold of concern or not. It could be 90 percent, but if the overall probability were so low, you might accept it. It could be 10 percent and not be acceptable.

There is no explicit answer. The point I would like to make is that many people don't even look at it.

MR. LEE: Could you also address whether you can expect to, or we should try to improve on our understanding of that overlap region?

MR. BERNERO: Absolutely.

MR. LEE: How much should we try to improve upon?

MR. BERNERO: That is the very thing we are doing. We do not consider it acceptable to look at the most likely or central estimate of the containment failure pressure and the containment load pressure and say, voila, it doesn't fail, or yes, it does. That is not an acceptable analysis, in our view.

We have to look at the tails insofar as we can construct them, but that requires a proper treatment of both load pressure and failure pressure.

MR. LEE: Do you think we can get reasonable estimates of that overlapping region by sometimes next year, before we make a decision on --

MR. BERNERO: Yes, I think we can.

MR. LEE: I would like to also ask you one more question.

At our meeting with IDCOR people yesterday, it was brought up by the IDCOR people that perhaps the early containment failures are not as likely as they were thought some years back. Hence, that fact alone perhaps could change a lot in their deliberation or decision process toward perhaps containment alterations, or venting, or whatever. And I got the distinct impression that they were talking about hours versus minutes.

Could you comment a little bit on whether these early containment failures are not as likely as they were thought to be?

MR. BERNERO: They are relying on analyses such as Indian Point and Zion where there is a containment failure pressure, a very robust, large containment, which is substantially above the containment load pressure, and therefore it is fairly obvious that the tails, whatever they are, are going to be small. The overlap potential is rather small. And on that basis they would say, huh, those containments are not likely to have an early failure. However, they are also referring to some of the earlier risk assessments, which gave a strong weight to early containment failure, by superimposition of loads, hydrogen loads, and steam loads together, and so forth.

The IDCOR people, in their analysis, are looking to establish a large gap between these two pressures. I have yet to see any analysis by them that amounts to a convolution of uncertainties to see what the overlap might be. But I do know that they are looking at both containment loading and containment failure pressures in a more rigorous manner than previously. But I haven't seen any results yet that indicate that they are as deep a look as I would like to see.

MR. LEE: If I may come back to my earlier question: You said that you would like to accomplish a better resolution of this overlapping region and so on sometime by mid next year or so on. So as a part of such an effort, you have been working on RETAIN code and things like that, as well. But do you feel that we are in a position with the code where you can perhaps try to predict, or try to -- yes, I guess, predict the procedure that will be required for the information regarding, for example, penetration probabilities, and things like that?

MR. BERNERO: Yes. I think so. In the next presentation I'll be showing you a schedule of what we call the source term reports and analysis and the review thereof that I think will give you a sense of the scale and the timing of when we are trying to make these appraisals, and we are using a set of five plants there that leads into the severe accident set of more than that. And I think it will give you an idea when we think we can make that decision.

MR. KERR: Do you expect to get that containment failure pressure distribution -- or get better information on it from analytical work, experimental work, expert judgment, all of the above?

MR. BERNERO: Yes. All of the above, really.

If there was one program that -- well, frequently, when I look at experimental programs, I say, gee, if there was one program I would accelerate, this is one. The containment experimental program, and I'll just call it penetration qualification program, you know, penetration research -- I would really like to see those done yesterday, you know, much earlier. I wish that were so. It isn't so. All we can do is look at the results as we go along and see if we can't extrapolate or forecast what the outcome will be.

We will have to rely a good deal in this regime on expert judgment as much or more than experimental data, because the programs just run over a longer scale, which the time scale is several years. So there is some data just coming in now,and a lot of it is empirical data, analyses of previous containment tests, containment experience.

MR. KERR: Back in the days when pressurized thermal shock wasn't being looked at so much but people were still concerned about reactor pressure vessel failure, a lot of operating experience on pressure vessels that was nonnuclear was used to draw some inferences about the behavior of reactor pressure vessels.

Is it possible to make use of existing information on the behavior of conventional vessels to get more information on that, or are they so different that one can't draw conclusions from them, or do the data exist, or whatever?

MR. BERNERO: Bill Farmer, on the penetrations, could you say that nonnuclear penetration data is of significant value?

MR. FARMER: I'm not aware of any nonnuclear electrical penetration leak rate data, and we have very little to go on when it comes to leak rates.

MR. KERR: Most of the uncertainty you feel, then, is in the penetration?

MR. BERNERO: Yes. We generally speak of penetrations and seals, things like that. It is that mixture of peculiar gaskets, diaphragm seals, some of the big butterfly seats on elastomeric donuts instead of walls. So it is those details.

MR. KERR: So you are talking about things for which data wouldn't be of much use, anyway? Think of the data you would have to have to cover the different plants. I mean, the statistics --

MR. BERNERO: Well, it's plant-to-plant differences.

MR. KERR: Well, differences with age of a given plant. It seems to me it is hopeless to get the data from experiments.

MR. CORRADINI: I follow that up, then. The question has been partly asked already.

If you've got two curves, and I want to look at the region in between, or as you said to John, that the region in between will have a fairly good handle within the next year or so, which of the two curves are you working on to get the handle on?

MR. BERNERO: Both. For instance, this tail here is a particle-size tail, principally.

MR. CORRADINI: Principally?

MR. BERNERO: Ha, you wince. And you know more about it than anybody else. Now I'm going to pick your brain.

You know, the question, how well do you have to know it cannot be answered in advance. You have to take how well you know it and examine the result after you look at it to see whether you can make a sufficient decision.

Now, obviously if it turned out that for the vast majority of the plant these two curves were almost congruent, you would have to know both of them with consummate precision.

MR. KERR: You just made a statement which I can't let pass without exploring.

I don't see why you can't make some estimate of how well you would like to know it in an engineering situation. It may turn out you can't know it that well, but given the uncertainties that you are willing to accept to operate with, it seems to me that one could make some estimate of how well this is needed to be known.

MR. CORRADINI: If you ask the question in reverse, maybe it would tell you what you are looking for.

MR. BERNERO: We are doing just that. We are taking large dry containments first. As an example, we are looking at the large dry containments for their pressure volume capability on the books, multiplying it appropriately on the advice of the structural engineering community to get the central estimate of 2.2 to 2.5 times design pressure, which becomes the central estimate, and then looking at the containment loading sensitivity study, trying to find its central estimate, to see how far apart they are.

We are just trying to establish that distribution from which will come an index of how well we have to know it. The reason I raise the 10 percent was we took one fast cut through the large dry containments and got a potentially significant overlap, enough to tell us go do that homework deeper.

MR. KEER: Maybe I can get Mr. Siess to explain later, but I'm puzzled if you think the failure is most likely to occur in containment, that the best central estimate is that based on structural considerations?

MR. SIESS: I didn't know. I guess I was going to ask, are you doing a similar kind of study on the probability that there will be a leak rate in excess of 100 percent a day? I assume that failure here still means gross catastrophic rupture.

MR. BERNERO: Yes. As I said earlier, we are following two approaches: the threshold model and the leakage model, like the curve I drew on the board. And in reality, as you know, this tail is the leakage curve. You know, if you really knew perfectly what was happening --

MR. SIESS: Not the way you are drawing it. That is a structural tail you've got there.

MR. BERNERO: That is artist's liberty, just to show that there is overlap. The shape of that distribution would be such that in reality the leakage model brings you up to here, and the tail going beyond is the tail associated with the uncertainty about the asymptote.

MR. SIESS: If you drew it, instead of leakage, on the size of the hole, the equivalent size, the pressure, what you would be plotting here would be the probability that you would get a great big hole?

MR. BERNERO: This is probability of a risk-significant hole.

MR. SIESS: That is the probability of a great big hole.

MR. BERNERO: Yes. This is more the probability of a catastrophic failure.

MR. SIESS: Right. There's another curve that is a probability of a four-inch hole, or a half-inch hole, or a ten-inch hole, which has nothing to do with this curve.

MR. BERNERO: It would be a different curve. This is a threshold model that says there is a threshold at which the containment fails and below which the containment does not fail.

MR. SIESS: The trouble with that terminology is that the threshold is at the extreme upper end; it is not the kind of threshold we normally think about; this is our threshold model. It may turn out that there is no way of ever getting to this so-called threshold.

MR. BERNERO: It may. That is why we are equally pursuing the leakage model which may render this whole consideration useless.

MR. LEE: But if you superimpose this leak probability curve with the membrane failure probability curve, perhaps, you may still see some general distribution of the type that you are showing, perhaps.

MR. BERNERO: You may or may not.

MR. SIESS: You can't do it in this format, no, because the size of the hole -- at certain levels, the size of the hole may be a function of containment pressure.

MR. BERNERO: This particular presentation has as a separate parameter not displayed the size of the leak. This is for any given size of leak.

MR. SIESS: The way you've got that drawn, it is for a great big leak.

MR. BERNERO: It is nominally for the great big leak.

MR. SIESS: In your threshold case there is either zero leak or a great big one. That is what you mean by threshold.

MR. KERR: Now that we have solved that problem, let's continue.

MR. BERNERO: Now let's talk source terms.

You may recall that in December of 1982, which was last year -- I'm losing track -- there was formed a group called -- no. Wait a minute. Back up.

December 17, 1982, there was a memo from office directors to the EDO with an action plan for accident source term. That is the sequence of events which said, let's get out and get source term related research information and get it into the regulatory process in a nice, quick, timely and responsible manner. And it laid out milestones that included, as early as February of 1983, initial assessments of source terms, what might be called interim source terms. But a lot of activity in 1983, somewhat culiminating in a September '83 approach to the

Commission, with revised policies, standards, or something or other related to emergency response planning, based on revised source term knowledge.

Then in January of 1983, the EDO established a separate management group called the Accident Source Term Program Office, ATSPO, and I was named to be the director of that Source Term Program Office. It is in the Office of Research, and with but one exception is staffed by people on detail from appropriate sections of the Offices of NRR, the Office of IE, and the Office of Research.

And its basic charter is to develop or see to the development of source term information, accident source term information, and work that into the regulatory process in a timely way.

Now, the basic agenda, as I have here -- remember, the prospective of this office, this ATSPO, is to do what needs to be done in the coming year or so, you know, give or take; document the current data base for severe accident behavior prediction.

We have a moving target here. There is an awful lot of work that has been done and that is being done, and it calls for a current documentation, a snapshot. It is time for another NUREG 0772, which is a state-of-the-art appraisal for accident source term prediction.

Secondly, the agenda includes the application of the latest techniques or the latest knowledge to estimate severe accident source terms for at least some reactors, at least a representative set.

Thirdly, the agenda must include, obtain, be substantial and broad peer review.

MR. KERR: Excuse me, Bob. What is the significance of the term "latest best estimate models"? Does that imply that one is developing some new models, or that you take the new data and on the basis of the new data try to put them into the old scenarios?

MR. BERNERO: It is both. It is developing the most realistic, and that is why I use the term "best estimate," rather than relying on repeated use of MARCH and CORRAL, to model, based on new research information.

It is to develop more realistic models, perhaps more difficult, more elegant, more deeply analytical, of the processes within the reactor coolant system in particular, and to develop those models using this data base.

The peer review has to have two compo... to it. For one, when you are going into this regime and trying to get best-estimate models for the physical processes of core melt, physical product transfer, all of these things are extremely difficult and require the best advice of all of those people who are involved in this kind of work and have been involved in it in recent years. And as a result, there is a very, very strong need for specialist review, for the specialists in these Archean sciences.

In addition, there is a very important need for, I would call them, the scientists across the street. There is a need for a detached, not specialist review, to look at the basic level of science, to look at the forest rather than the trees and to ensure that the basic science that will be used here is sufficiently well-grounded.

Because, if you believe much of what you see in industry papers and pronouncements, this source term estimate that is now very popular is such that, well, you know, we used to say that reactor accidents couldn't happen. But now, unfortunately, we do have to admit they do happen; but son of a gun, under close scrutiny, they don't hurt anybody, nothing gets out. You know, the forces of nature hold everything in, and isn't that wonderful, and if it is true, let's prove it, and let's prove it responsibly. If we are going to base regulatory action, regulatory decision on anything like this, we better have a sound scientific basis and a broadly accepted one, and unless we get both deep specialists and broad peer review of the principal work here, we won't be able to use it.

Now, we also have to look in the first order of business inthis agenda at emergency planning, and in particular we have had about five years -- if I could recall for your memory, emergency planning is one of the few risk-based things in reactor relation. Prior to 1978 there wasn't a whole lot done on emergency response planning. It was there, but it really wasn't very well fleshed out. And during the period of 1977 to 1978, the Environmental Protection Agency and the NRC worked together to develop an important trail of reactor risks suitable to be the basis for emergency planning. That was published in December of 1978 in a document, NUREG 0396, which also had an EPA number. It was a joint report by the NRC and EPA. And what that report did is, it took the risk models of WASH 1400, translated them into risk versus distance, different relationships of the probability of suffering a certain dose at a certain distance from a reactor. Again, WASH 1400 model. And that report concluded that, based on this portrayal of reactor risk, the level of probabilities and the reach, the range of life-threatening doses, considering the protective action guides that the EPA had out in draft then, you know, the 1 rem, 5 rem, and so forth, protective action guides; that considering all that it was appropriate around nuclear power reactors in this country to have a 10-mile radius emergency planning zone for the immersion pathway, that is, cloud exposure of humans, and a 50-mile radius planning zone for the food chain pathway.

That document went on the street and became the basis for emergency planning in late 1978. Many people don't realize that when the TMI accident happened just a few months later, and then there was a great deal of interest in accelerated pursuit of refinements in emergency planning, that document stood as the planning basis and the derivative document jointly prepared by NRC and FEMA, the Federal Emergency Management Agency, NUREG 0654, that came out; and it actually came out in a revision later, and I forget the dates, but early 1980 and late 1980, something like that, and the document NUREG 0654 establishes the criteria for such emergency planning. So you have one as the technical basis, the other is the criteria.

Obviously, if you reevaluate accident source terms you go at the very root of the whole structure, and you have to go back to NUREG 0396 and say, is that still a fair portrayal of risk.

Is the conclusion, the recommendation of a 10-mile planning zone for immersion pathway, or a 50-mile planning zone, a logical one. You also have to go back and look at what we have done in the five years since 1978 to implement that portrayal of risk, that basis of planning, and those criteria.

Have we learned something from that implementation such that we would reconsider it even without a new source term?

So you have to look at the experience as well as the technical basis.

So, we have to go back there, look at this planning experience while we are looking at the potential revision of the accident source term, and then given new source term information and refined understanding from the evaluation of experience, our agenda is to go first into the emergency response regime and develop revised policies or revised criteria, and then later to go into things like the equipment qualification source term; what is the appropriate one to use for different equipment?

Design bases for water cleanup systems, and things like that. Thos would be later. And, of course, the severe accident decision is one of the others.

MR. KERR: Even though you might use the same technical approach that you used in 0396, it isn't obvious to me that you would necessarily change the 10-mile zone just because the source term changed.

MR. BERNERO: No. You might say I'm still planning for 10 miles, but my skew of planning, my distribution of attention is going to be different. This is, quite frankly, one of the things we are discussing. I might approach the first two miles as zone 1 of the 10 miles, and that is the one where life-threatening doses are most significant. Zone 2 would be from 2 miles to 5 miles, a secondary threat of life-threatening dose, or urgency, and then another zone from 5 miles to 10 miles. I could grade my attention, and I might leave the alerting systems and the organizational structure alone and still work with a 10-mile planning zone and just absorb the conservatism.

On the other hand, if the source term reduction were really dramatic, I might just say, oh, I'm going to cut down from 10 miles to 5 miles, or to 3 miles or something. But the option is open.

Now, our approach in this work is to identify these elements, and they will turn out as you see to -- wait a minute. I skipped one. Here is our strategy.

The first thing is to do a reassessment of the source terms. NUREG 0772 was really a snapshot in 1981. It said here is what we know about severe accident source terms and here is what we would say about predictions. And if I could very simply state its conclusions. NUREG 0772 two years ago said WASH 1400 estimate is far more likely to be conservative than it is to be optimistic. It said the phenomena are such that the releases from severe accidents, that is, core melt accidents, are probably lower than this but we can't establish yet how much lower; we can't establish a scientific basis yet for a lower number. It's promising; we've got to work on this, but we are not there yet.

MR. SHEWMON: I wish you would use a different word than "optimistic" as the opposite of conservative. Realistic might be a better word, I hear behind me here. Conservative means we won't get there, where optimistically --

MR. BERNERO: It is really overestimating or underestimating. Does WASH 1400 accurately estimate, underestimate, or overestimate. And what NUREG 0772 says is that WASH 1400 is likely to be an overestimate of fission product release.

MR. SHEWMON: Normally I like to be optimistic, and I get confused.

MR. BERNERO: Now, our research program, if you look at it, in the ideal, says be patient, friend; 1981 is nice; you stick around and in 1985 we will tell you a whole bunch of new things, because that is when we get the best fuel damage data, the best codes, and the best all kinds of things.

MR. KERR: Now, in planning your source term work, have you sat down and asked yourself what sort of information do I need if I'm going to make a significant change in emergency planning zones, for example? What are the sequences, or what are the isotopes about which I need most information, and what sort of information do I need?

MR. BERNERO: Well, yes, we have. This was done quite some time ago, in fact. We have the isotopes broken into the categories of relative significance, like the radioisotopes, which are very important, for the early fatality or early radiation injury and thyroid doses. The iodines, iodine and antimony have to be treated separately. Tellurium is one that is quite significant in dose and quite spooky in its physical behavior, difficult to predict. That was identified a long time ago. And we have reestablished those identifications and called for the physical research that would illuminate just what those nuclides in particular do.

Then, of course, there is the basic need. Nuclides like cesium, cesium 137, dominates latent cancer risk.

MR. KERR: What sort of changes in source terms would lead to changes in emergency planning?

MR. BERNERO: What we have done and have reported, the best source, if you want to see some of this, is in the siting study. We took in the siting study -- there is a whole chapter dedicated to this. What reductions in source term will produce what changes in risk. There is a parametric treatment, a factor of 2, factor of 5.

MR. KERR: But given that, you have to decide how much of a risk reduction is going to lead to a change in emergency planning? If you get a change in risk of 10 percent, you probably won't do anything.

MR. BERNERO: That is the reason for the sensitivity study. In a nutshell, what it shows is that if you can get even a factor of 2 or 3 reduction, well-established for nuclides like iodine, you have a dramatic effect on the immersion pathway fatalities, early facility doses, a dramatic effect. If you look at the sensitivity studies that have been done --

MR. KERR: Does that mean you reduce the planning radius from 10 miles to 5 miles, or 2 miles?

MR. BERNERO: There is often a great deal of argument about why you have emergency planning. The emergency response planning is basically to handle radiation doses with predictable effects. If you get doses above 50 R, you get clinically detectable radiation effects.

Your hair falls out, you vomit, you start suffering radiation injury, and of course if you get over about 2 or 300 R, you start getting early fatality, prompt fatality, and radiation planning has two objectives. One is to avoid exposure to minimize the probability of exposure to damaging effects, to known radiation effects, and it has a secondary consideration that is influenced somewhat by stoichpastic effects like latent cancer, and that is to minimize the probability of people suffering doses in excess of the protective action guides.

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If you reduce the source term overall and you put a rheostat on everything and turned it down, the effect on early fatalities would be coming down something like a square log. That is, for a two-fold reduction of source term, you would get about a four-fold reduction of early fatalities.

At the other end of the spectrum, latent cancer exposures, cesium-137 is only going to come down roughly linear.

You are going to have a new perspective when you look at that ten miles. Your emphasis or use of emergency planning up to ten miles is going to be much more toward the protective action guide exposures than to the life-threatening doses.

MR. KERR: I thought you told me that reduction of a factor of 2 would have a dramatic effect, and I was trying to understand what dramatic effect implied in terms of emergency planning. Does it imply that you would do something dramatically different in emergency planning?

MR. BERNERO: A factor of 2 could drop the fatality radius quite a bit. A factor of 5 or 10 could bring it in very, very close to the reactor. You are dealing probabilistically.

MR. KERR: What I am trying to get at --

MR. BERNERO: You could possibly get to the point where prompt evaluation would be a method of choice, an joy2

emergency response method of choice only within one or two months. You could get there. Anything beyond that would be --

MR. KERR: How much reduction in the source term is likely to lead to that conclusion?

MR. BERNERO: Five to ten, I would say. Something in the range of a factor of 5 to 10 overall would lead to that.

MR. KERR: Thank you.

MR. BERNERO: As I was saying, NUREG-0772 two years ago gave us a nice snapshot. We would like to wait until 1985 for the next snapshot, but that would not be timely; so we have scheduled an interim snapshot and given it an interim number, NUREG-0956, that would come out in 1985 or at the end of 1983, and we will say we are not finished with the work yet but here is a reassessment of the technical basis for estimating fission product behavior in severe accidents.

Now, while we are doing that -- in other words, while the physical chemists are doing the best they can to give us this information, we have in parallel closely been observing a reevaluation of the various relationships of risk, sensitivity analyses, significance analyses, and a reevaluation of the emergency response experience.

You know, we have had a lot of drills, a lot of practice exercise, a lot of regulatory experience, both our

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agency and the Federal Emergency Management Agency, and all the state and local parties involved.

So, with these two things going in parallel, then essentially at the end of 1983 they come together in the first step --

MR. KERR: Excuse me. I don't understand how the current assessment and the emergency response experience is related to the source term. Give me some information on that.

MR. BERNERO: Well again, I would recommend that you read NUREG CR-2239, the so-called Sandia Siting Study, because what we did is we took all the sites in the country with WASH-1400 source terms, simplified and re-baselined, still WASH-1400, no new reduced iodine stuff in it, and we simplified them to three core melt accident types: a very bad core melt, a middle kind of core melt, and if you could call it one, a nice core melt. You know, different grades of emergency safety feature availability, one, two and three. And what you will see is there is a systematic analysis of the risk relationships associated with each of them: how far out are people at early death risk, how far out are the latent cancers and so forth, what is the distribution of doses and what is the sensitivity of each of those things to changes in source term, and other parameters like population and so forth.

So the purpose of that study -- that is really not

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merely siting.

MR. KERR: I'm trying to get to the relationship to assessment of emergency response experience. To me that means you take the -- I thought it meant you took these practices in --

MR. BERNERO: Practices and evaluations in any model here. Bill --

MR. KERR: I'm not disagreeing with you. I don't understand what it is you are comparing with what.

MR. BERNERO: For years we have modeled the risk by making assumptions about evacuation.

MR. KERR: So you used the experience to refine your predictions of evacuation behavior.

MR. BERNERO: The decision times, how soon can decisions be made, how effectively can people evacuate. If you look at NUREG-2239, there are sensitivity studies already available with WASH-1400 source terms which say what if it is a mixed bag instead of everybody goes promptly? What if you've got a 30 percent probability they are going to start leaving in an hour and a 40 percent probability that they will leave in three hours?

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MR. KERR: You can use these new data with existing source terms and have perhaps reached different conclusions?

MR. BERNERO: You could do this all by itself and reach different conclusions.

MR. KERR: I understand.

MR. SHEWMON: You talk about them leaving at different times. Have we given up completely on them going into the basement and staying there?

MR. BERNERO: No. These are mixes of sheltering evacuation; no protective action at all. Mixes of protective actions.

MR. SHEWMON: Thank you.

MR. BERNERO: It is that evaluation with existing source terms in light of five years' experience.

MR. KERR: Have you practiced having people going into their basements?

MR. BERNERO: Not that I know of.

MR. KERR: I'm not being facetious here. Have there been certain exercises in which you told people, "Go to your basement"?

MR. BERNERO: To my knowledge, there has never been an exercise that involved the public response other than the public agencies -- police, emergency response.

MR. SHEWMON: The instruction is to stay home, stay inside?

MR. BERNERO: Close your windows. A la TMI.

MR. SHEWMON: This is part of what the local authorities considered?

MR. BERNERO: Yes. As I say, you could do the evaluation with the existing source terms and try to reach a conclusion on it alone. Actually, with the two together, you know, it makes more sense to do the two in parallel, and then incorporate source term reassessment as available and appropriate and do that.

Now, we have in this work two types of peer review. I mentioned earlier there is the specialist review, and, in fact, the second one is scheduled for May 24th and 25th in this room. The Surry report was done in this room. There is a two-day session where the specialists get together, they read the detailed modeling and analysis and chop it all to pieces, and we have a very nice time. That is necessary. We have to do that to get the best state-of-the-art. But then we have arrangements for a broad, scientific review.

Right now, we're negotiating with the American Physicians Society to see if they might conduct one of their studies. You know, they have a history of doing technical studies of important scientific work, and they did an excellent study of the Reactor Safety Study. You know, the original Reactor Risk Analysis. It's not really the original; the 1973-74 one. And we're negotiating with them to see if they can provide specific, well-founded substantial effort to conduct a broad, scientific peer review.

MR. KERR: Is this because the American Physicists Societ does a better study, or because you think there's more physics or chemistry involved here?

MR. BERNERO: No. It's the former. It's heavily chemical. There's an awful lot of chemical work in here.

MR. KERR: The last I looked, there was the American Chemical Society.

MR. BERNERO: There's also the history that the American Physicists Society goes to the other technical societies and draws in expertise, and quite frankly, they have a history of finding study objectives and doing them promptly.

The National Academy of Sciences and other expert bodies are certainly expert, but when you look at the sequence of events -- and I'll show you in a schedule -- it would be nice if you could produce whoever they are to give you prompt feedback, to do good work and do it promptly.

MR. SHEWMON: Professor Kerr, there is also, in effect,a good physicist firmly convinced they can do anything if they set their mind to it, whereas,the good chemist may not quite have that conviction. I don't know.

MR. BERNERO: Chemical engineers are usually humble; physicists are not. There are a lot of other agencies involved. The ACRS, of course. There will be a good deal of reaction with you as time goes on, mostly in the arena of severe accident research and results,but because of the complex nature of emergency planning, we have a lot of contact with EPA, with FEMA, with the many state radiation authorities. You know, the radiation control program in each state is deeply involved in emergency planning. The various governors' offices and the like.

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MR. KERR: I think it is appropriate that ACRS should come right after "etc."

MR. BERNERO: I didn't get you. I didn't know whether to call you an agency or not.

Now, the elements of this work may be set in four. One is what is the data base we have for these predictions.

Element 2, is the set of estimates of accident source term, accidents for some selected plants, and accident sequences.

Now, notice I have said here selected plants and sequences rather than a summary risk appraisal, and I say that for administrative reasons. I put that in Element 2. I'll talke about Element 4 later to explain that.

Then a strong peer review of the preceding -- the scientific basis for reassessment. Some rather complex chemistry; heat transfer, mass transport, physical phenomena involved, and we have to have a reasonable degree of confidence, whatever the criticism we heard in the first one. The specialists' peer review of the Surry work in this room was replete with scientists saying your calculations are getting ahead of your science. You've got models; where's your validation? You've got great big tissue papers of analytical models and very little evidence that you have scientific data to evaluate those models. So we have a very strong check on that, that these two are matched.

And then, of course,we do have to appraise the risk and the regulatory significance of the reassessed source terms. It is in here that we come in with the real insight on the containment failure. This is where we have to come to grips with what are the dominant sequences, what is the probability distribution for this sequence against that sequence, for this containment failure mode per sequence against that containment failure mode. This is where we have to get the risk perspective,and given that we've got NUREG-0956 --

MR. KERR: Remind me again, when you talk about a source term here, you're talking about not what is available for release but what is released?

MR. BERNERO: The source term is the characterization of the release of radioactive material from containment, as result of a given accident sequence.

MR. SIESS: Bob, in Element 4 I think I know what risk means, but I'm not sure I know what regulatory significance means. Now, if you explained that when I was out of the room, forget about it.

MR. BERNERO: No, I didn't. How could I put it?

We were talking earlier about how much of a source term change is there. If,after agonizing reappraisal we came up with the WASH-1400 source term should be corrected by the following factors to give us the new NUREG-0956 source term and the following factors turn out to be on individual columns multiplied by .98, .82, .91, you know, why gild the lily? You are niggling with 10 percent changes or less. It's not a significant source term.

On the other hand, if you have 5 to 10 factor type changes, then the regulatory significance might be a commentary that says life-threatening doses now cannot reach -- with any credible probability,they cannot reach very far beyond, say, a mile, two miles, something like that. And the regulatory significance of that is potentially, a dramatic change in the plant.

MR. KERR: Chet, I would like to take that to Main. Can we get those guys in NRR to brief this PRA stuff?

MR. BERNERO: They believed it the first time. It's the basis of existing emergency planning.

So these four elements, having been performed,will constitute the NUREG-0956 effort. Now, I have a schedule here that we are working to,and you'll be seeing these things as a series of reports.

Element 1, which is the technical data base -- it does exist and will exist in many, many reports, but there is one particular report being pulled together by the Oak Ridge National Laboratory to summarize the technical data base, specifically directed towa.. validating these predictive models, these source term models.

Do we have a number for that report yet? I don't think so. It will be an Oak Ridge National Lab report, and that will state the technical data base with particular emphasis on validating these codes.

Now,at the same time, the Battelle-Columbus Lab --

MR. KERR: Excuse me. Is the Oak Ridge report Element 1?

MR. BERNERO: Yes, it is Element 1. And insofar as it embraces this vast interest in the data base and states it in a way understandable and directly pointed toward validating Element 2, which is what it is supposed to do.

Element 2 is Battelle-Columbus report, and we have a number but I didn't put it on the slide here. The master Battelle-Columbus report will have separate sections or volumes for each plant, and we started out to do four plants and are actually going to dofive plants. And the plants are Surry, Peach Bottom, Grand Gulf, Sequoyah and Zion. We added Zion at the suggestion of a good number of the reviewers.

Now,these reports document the detailed fission product release and transport analysis. What they do is they use the MARCH 2 point on code, started out with MARCH 1.1, and it was widely criticized. They are now using the MARCH code, and the new code, MERGE, for the heat transfer analysis for the thermal analysis for the heatup of the core, the melting of the core and the attendant heat behavior, thermal behavior, of surrounding metal. You know, like the upper plenum and nozzles and so forth.

The MARCH-MERGE codes basically calculate the rate of heat-up and core melt.

The CORSOR code is used to calculate the emissions of radionuclides, aerosols and whatever gases from the core during its degradation and melting.

MR. CORRADINI: Excuse me. The difference between MARCH and MERGE? I have it somewhere here.

MR. BERNERO: MERGE is the radiant heat transfer, and convective heat transfer in the reactor coolant system.

MR.CORRADINI: Okay.

MR. BERNERO: The CORSOR code describes the behavior of the radionuclides getting out of the core,sort of the emission from the core.

And then the TRAP-MELT code models the fission product transport, and deposition within the reactor coolant system boundary.

Then once the MARCH code has brought the corium outside of the coolant system, the CORECON code is used to model core-concrete interaction.

MR. CORRADINI: I'll ask a question inbetween where it came out and where it made it to the floor. What is used is the hot drop sub-routine to do the spikes?

MR. BERNERO: Yes.

MR. CORRADINI: Since that is important.

MR. BERNERO: Very. Once the core-concrete interaction is treated -- and mind you,there are aerosols. The NAUA-4 code, that is the West German code, that is used for the containment behavior. That is,aerosol behavior in containment.

Now, the emphasis in all these calculations is on the sequence of events leading to the release and transport of radionuclides and their behavior, their plateout, their deposition.

The emphasis in these reports is not on the last word of when containment fails. It some respects, you could say the results of these analyses give you source terms as a function of time where you reach in and tell it when the containment fails. They are not the final answer.

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MR. SIESS: How did you pick those?

MR. BERNERO: Those plants?

MR. SIESS: Are they all PWRs of Westinghouse?

MR. BERNERO: Surry, Sequoyah and Zion. They are Westinghouse.

MR. SIESS: They are a large dry to subatmospheric and a big --

MR. BERNERO: A Mark-1 and a Mark-3. They were picked for historical reasons as much as technical reasons. As you know, WASH-1400 looked at these. RSSMAP looked at Grand Gulf and Sequoyah, and everybody and his grandmother has looked at Zion. The data is available.

MR. SIESS: You said earlier the difference to the bottom head is not something that goes into the codes anyway; you just tell MARCH what it is you want.

MR. BERNERO: You tell MARCH what came down. You tell the hot drop routine, I think it is. Here is the answer.

MR. CORRADINI: These, except for Surry, are the in-core plant?

MR. BERNERO: Yes. They picked them for pretty much the same reasons we did, and we had a strong interest because, see, they are using the MAAP/RETAIN code series, and it was important to us to have those cross-section capabilities.

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MR. KERR: Is there somebody that is writing a code that will make certain that those two somehow agree on the results?

MR. BERNERO: No. There are efforts under way to cross-check the results, you know, to see whether the analysis of a given plant for a given accident sequence using MAAP/RETAIN would give you the same results as using all this, and if not, why not.

MR. KERR: I just assumed that that could be done with the code. It can't?

MR. CORRADINI: Could I ask one more question? The interesting thing is your output here is not failure, you said, but essentially the aerosol density in the atmosphere and its chemical composition as a function of time.

MR. BERNERO: Yes, in essence that is it. You still have to face the question of early containment failure as against late containment failure. You can almost read off what the late containment failure source term is, and the early containment failure you can also read off.

MR. CORRADINI: Now, if I can strip away the code and just ask some of the physics, in your CORSOR or whatever it is, is it essentially thermodynamics? You are monitoring the temperature in a local region and seeing what is going to volatilize and come off?

MR. BERNERO: Mel Silverberg.

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MR. SILVERBERG: Given a time history in a structure in MARCH, what CORSOR does is now take that temperature and provide a release of more or less fission products as well as structural materials. That would come from empirical data from the SASHA work in Germany as well as the Oak Ridge work.

MR. CORRADINI: I'm trying to relate this to in-core because of what we heard yesterday. In terms of what the IDCOR people said, it would essentially parallel the calculations they are doing in one of their technical reports in terms of core fission products in their plant.

MR. BERNERO: Yes.

MR. CORRADINI: It gets into the lower plenum and it gets out of the vessel, and before it makes it to the concrete. What are you looking at in terms of -- what is the physics of how aerosols or fission products are released in that transition period between in the core and on the floor?

MR. SILVERBERG: That is omitted. There was some discussion in the Surry report about possible ways of aerosolization, if you will, during that step, but that step was not included quantitatively. The next step after it hits the floor -- we had that routine -- it's called the VANESSA routine -- which Sandia put together.

MR. CORRADINI: This is taking the superficial

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gas velocities from CORECON and getting in the lower area.

MR. BERNERO: Given that we have now --

MR. KERR: Aerosols are not generated until the stuff hits the floor?

MR. CORRADINI: The movement from core to floor, there are no aerosols right now being released.

MR. SILVERBERG: It is generated in the core while the core is heating up, as it should, but not during movement to the floor.

MR. KERR: Thank you.

MR. SHEWMON: They are certainly generated inside the pressure vessel when it melts, and we are mostly interested when they come out, and they don't certainly come out with these basketball-sized blobs that Bernero was shooting out through the bottom. They are coming out someplace else that we are interested in.

MR. SILVERBERG: The aerosol release is continuous according to the temperature, and the release rate is based on that data throughout the core heatup process. It is during the time when the MARCH code then allows it to -- assumes that it drops to the floor. During that short time step there is no, if you will, aerosol release.

MR. SHEWMON: But the aerosols then will have been streaming up out through by whatever is involved.

MR. SILVERBERG: That is right. The aerosols will

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have been following the steam flow and gas flow and be deposited according to whatever the physical properties are.

MR. CORRADINI: The question by Dr. Shewmon raises another one in my mind. If one wanted to do that, not saying that we should, heaven forbid, but if one wanted to do that in terms of the movement from the core to the floor, in terms of source term, is it important in terms of timing that one consider that, or is it irrelevant because the time integral of what would be released would be released anyway if it ended up on the floor?

MR. BERNERO: I think the latter is true, a short time.

MR. CORRADINI: Unless it would be a position by some that you bypass the floor completely, that you don't get into a core-concrete arrangement.

MR. BERNERO: Right.

MR. CORRADINI: Okay.

MR. BERNERO: With element one, the Oak Ridge report summarizing the data base for these predictions, element two, a set of five-plant predictions reasonably spanning the types of plants out there, the existing plants. Then we need the peer reviews, and I have here a date that needs an explanation. I put down December 15th.

By that time, we believe that we would have completed the specialist reviews of each one of these

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reports. We believe we would have had the advantage of the IDCOR studies. They are scheduled to be published in the summer, and we will see the reports and see the analyses and have a chance to go to our contractors and really analyze that, and we hope we can at least have a prognosis, I might call it a prospect, a qualitative indication out of whoever it is, the American Physical Society or others, that would say just in the broadest-brush treatment, your scientific work is generally sound or it's terrible or whatever, enough to know whether we could continue and complete -- it would be an opportunity for a signal not to complete the risk and regulatory significance work which is going on all this year.

This is for containment failure accident sequence probability and the like. Come to a head to say what do we know about risk today.

NUREG-0596 would be a reassessment of the technical basis for accident source terms using these five plants as surrogates for the class.

This information would then be put together in NUREG-0956, and in all likelihood this would be published as a draft for comment. You know, this is a classic example of the sort of thing where we have a fairly comprehensive body of physical information and analysis that really warrants putting it out for a public comment period.

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So approximately in February of 1984 we would be in a position to go to the Commission and say, here is a draft NUREG for comment that has got this great array of technical work in it, and here for now is what we think ought to be done by the Commission with respect to policy or plans or whatever, in emergency response or any other immediate conclusion that might be worthwhile.

MR. LEE: Where would I see the comparison of the results you get with the code with the experimental data base that you have documented in Element 1?

MR. BERNERO: Where would you see the comparison? I'm not sure what you mean.

MR. LEE: There would be some integral test as well as components of data base.

MR. SILVERBERG: It will be in Element 1.

MR. BERNERO: These are plant analyses. The validation of those codes and subelements of the code would be in Element 1 in the Oak Ridge report.

MR. SIESS: Bob, where do I see int raction with IDCOR?

MR. BERNERO: Well, it's going to appear down here. These things are produced on or about the same time that the IDCOR is produced.

If there's a substantial different between IDCOR results and the results we have, it has to be addressed in

Element 4. This is the sensitivity analysis. There is an uncertainty sensitivity analysis in there,too.

We can only rest on what they say. At this point, their schedule is to produce their reports on the same time-frame as we are producing these; in parallel, in other words.

At first, we exchanged a lot of information, but we are at a point now where we have to go in our separate rooms, and they are very close to the investigation with their work, and as a result of what is going to happen, these reports will hit the street roughly at the same time, and then the whole world can sit there and compare. And as part of our uncertainty and sensitivity analysis,we have to look at that large data dump from the IDCOR program and see what it tells us about the validity and quality of what we have done, or what our contractors have done.

Our expectation is that we will publish a draft NUREG-0956 approximately in February 1984 and that that would be really two things. It would be an opportunity for immediate action in the frame of emergency response if warranted, and secondly, it would be the documentation and a direct basis for severe accident decisionmaking that is available -- so-called 084 decision. Because it is this very work that extended to other plants as well, not merely the spectrum of five, but extended to other plants as well and would be the basis for a 1984 decision about severe accident mitigation or prevention in the population of existing plants.

So that is where we stand on the source term. I'd like to turn over the floor now for the discussion about flooding that Charlie Kelber will discuss. Is that okay with you?

MR. KERR: Are there anymore questions of Mr. Bernero?

We thank you for a very succinct presentation.

MR. BERNERO: I've never been succinct in my life.

MR. KERR: Charles, if you don't object, I'm going to suggest about a 10-minute break before your presentation.

(A short recess was taken.)

MR. KERR: Mr. Kelber, are you set?

Mr. Shewmon, I don't want you to miss a word of this.

MR. KELBER: I'll be as brief as I can. I have three photographs,and I think I can go through them fairly fast.

The paper called "Improvements of PWR Plant Responses to Severe Hypothetical Accidents" is the French national position on this topic and gives a detailed exposition of limited material handed to you last month during your meeting with the Group Permanente.

I would like to summarize the approach that they take to severe accidents in general and address directly precisely what they mean by flooding and what we mean by flooding.

In the first place, the CEA and EDF place a maximum of reliance on prevention with the stated criterion of a likelihood of 10^{-6} per year for a severe accident. They have, in their 900 megawatt and 1300 megawatt plants -- and I believe, also, in the N-4 plants which will not start construction until, I believe,next year -- installed instruments at several points in theprimary and secondary coolant systems to measure the temperatures, pressures and flow. These are over and above the ordinary instruments needed for either the traditional safety or control measures.

Those data are analyzed to produce an indication of one or more of 36 states of varying seriousness for the system. Six for the primary and six for the secondary. This combination is actually physically reflected in a matrix, six by six matrix, of lights that is in an indicator panel on tne wall of the control room at the N-3 plant. The first of which I saw was at Palwell.

Corresponding to those indications, procedures are prescribed to the operator. In developing these procedures and doing the analyses of the data, no significant number of multiple failures was assumed.

There is a very high reliance on the concept of a standardized plant, so there would be simply one analysis done for all plants of a given type. There will be one form of training; in fact, a simulator will, I believe, be physically located in Paris but remotely accessible.

We are negotiating for details of these procedures.

We've had many conversations with the staff, but we have, in fact, received no reports, though I believe that some may be forthcoming.

Now, in the discussion of what do you do if these procedures are not sufficient, EDF did, in fact, consider flooding the basemat drains. I should explain that this is applicable to a 1300 megawatt plant, and in correspondence to a containment configuration we do not have.

MR. SHEWMON: I had the impression that they were interested in getting water into the pressure vessel or the sump or several things before and had worried about that in addition to the drains.

MR. KELBER: They may have, but it's not within the context of national position, and I'll address some of the concerns that would arise out of that because we, too, have thought of that.

But let me say in the paper handed out to you, there is this diagram and I have included an enlarged version.

This is the configuration of the N-3 plant containment, which is a double concrete shell type of containment with no stainless steel liner on the internal containment. Because of that, it is expected that water will diffuse through the shell and drains have been installed leading to an access gallery, the gallery for inspection and possibly replacement of pre-stressed cables, and thence to be removed via the normal waste removal system.

Now, the failure mode anticipated by the French for both their 900 megawatt and N-3 plants in the case of severe accidents is the slow over-pressure and failure by basemat melt-through.

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The failure in the N-3 plants arises when the molten material reaches the level of the drains. The proposal was put forward by EDF, and I haven't seen any analysis that these drains then be flooded through this access hole and the gallery in order to retard the penetration into this much thinner basemat shell below.

Without knowing the details of the configuration, I would not want to comment on the feasibility of the scheme. I believe that one would be concerned about vapor blocking, depending, as I say, on the details of the geometry, and as we learn more, we will, of course,pass that on to you.

This is the only discussion of flooding that is in the French national position. However, I think that a number of groups have discussed flooding -- certainly we have -- and let me summarize my presentation by relating to our considerations, which I believe are no different than those that anybody else would make.

First, there was an internal study, a very simple one, done in 1977 to determine if the vessel might crack, and the answer was no. In the original Zion-Indian Point study done some years ago, this flooding was identified as an attractive option for mitigating the results of a core melt accident.

MR. SHEWMON: When you say flooding, you are talking about flooding the containment?

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MR. KELBER: Yes. We do not have the basemat configuration as in the French plants, but there are some pitfalls. First, the molten fuel will continue to enter the basemat even after the water covers the fuel. We have observed this experimentally just from the analysis of the thermal conduction and the heat transfer modes. There is no reason to expect that it would be anything otherwise.

On the other hand, the water, if deep enough -- and I'm not sure just how deep is deep enough, but I would imagine of the order of several feet -- will be sufficient to scrub some fission products that will be released by the interaction, and certainly it will remove some heat.

In other words, it will tend to act as any suppression pool would tend to act.

One of the pitfalls is that if the pressure does build up high enough from the steam, the hydrogen and the other noncondensable gases released into the containment, the water, which is highly contaminated by fission products, may be forced backwards through the connection to the outside, and you would then open up a direct pathway to the outside, and not only that, you would release a large number of water-soluble fission products into basically an uncontrolled area.

Also, you will have --

MR. SHEWMON: Let's talk about why the pressure

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would build up.

Let me talk for a minute, will you?

Are you postulating here, for example, that the containment spray is not operating, or are you thinking of a very high speed buildup which would come only with some particular local catastrophe?

MR. KELBER: I am assuming that the containment spray is not operating. When the containment spray is operating, most scenarios we see, it is very, very long periods of time before the pressure is built up high.

I would assume, then, where the containment spray is not operating somewheres within 100 -- well, somewhere within 6 to 10 hours, the pressure should get really quite high.

MR. SHEWMON: If the containment sprays aren't operating, is it because we don't have a pump around that is working, or --

MR. KELBER: The favorite scenario is then TMLB scenario, which is the loss of power.

MR. SHEWMON: That means you cannot drive the fire engine up and put it in because we haven't designed an outside stand pipe or because the plant is inaccessible or we just haven't worried about it.

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MR. KELBER: We haven't gotten to that detail.

I think there would be significant virtue in consideration of this problem to consider that. However, I think I should point out that the sprays themselves will still have to operate against a fairly high backpressure, and this means you have to have a pump capable of overcoming that backpressure.

MR.SHEWMON: Overcoming a one atmosphere backpressure?

MR. KELBER: Overcoming the several backpressures that might accumulate in the containment.

MR. SHEWMON: Several atmospheres, you've got other problems with the containment.

It seems to me the main thrust of what interested me was that as I understood the French, they were looking into operations of getting water into the containment at that time, and that's why I'm hoping that maybe we might be, too.

MR. KELBER: We are, and I'm pointing out all the considerations that have to be made. Every one of these schemes has its pluses and minuses.

Excuse me. Now I would like to finish. What I pointed out is that the timing of introduction is a substantial question here, because if you introduce the water at too late a stage, you have to guarantee that you have sufficient head in the pump to overcome the pressure that exists within the containment, if I understand my hydrostatic analyses correctly.

Now, that is simply a design problem that has to be resolved and it brings with it certain penalties and certain problems that have to be faced. It's not a simple-minded matter. But we have considered it and we are considering it.

MR.SHEWMON: Staying with that consideration --

MR. KERR: It's my time. The low pressure injection system, in a sense, has its problems. It has to inject --

MR. KELBER: Yes, of course.

MR. KERR: That doesn't seem to be an insurmountable design problem.

MR. KELBER: In the analysis of the TMLB' accident, for example, one finds that one draft pressure in the system -- that it is high for a very long time. Finally, after vessel melt-through the pressure which has been contained within the primary system is suddenly detained within the containment. I believe this is nominally 600 psi.

Now, that causes the accumulators to dump, and they dump out to the molten core, creating all sorts of problems.

The steam thus generated plus the hydrogen tends to build up the pressure, and depending upon the model you choose, within a few minutes to a few hours, the pressure is now back up to several atmospheres, depending on the size of the containment. And at that point, you now have to ask, what are you going to use to pump the water in with.

Well, you're going to have to pump it in with the pressure exceeding the pressure in the containment which may be close to 100 psi. It's a simple problem but it brings with it certain consideration that says you have to have now in isolation two sets of isolation valves that have to be opened against this pressure and allow you to net water at a somewhat higher pressure. How reliable are such valves? I don't know.

MR. KERR: I think the low pressure injection system that is used has precisely these requirements. You have to have isolation valves against normal system pressure,and you have to operate -- these pumps typically are capable of supplying water against a head of 200 or 300 psi.

MR. KELBER: I don't believe they're double isolation valves. If they are, you may want to use the same system.

MR. KERR: There are two check valves, in series.

MR. KELBER: Those who want to put a lot of reliance on check valves are free to do so.

MR. KERR: I'ms simply saying that is now done.

MR. KELBER: You wouldn't use the check valves in this system because they would normally be opened to the outside.

MR. KERR: I don't see -- well, let's not design it at this point. It isn't clear to me that it's an insoluble problem.

MR. KELBER: No one said it's an insoluble problem. It raises problems of reliability and mode of operation.

Finally, you've opened up another possibility for loss of isolation, and you have pretty much committed to having a wet sump, and I think that question is still open as to whether a wet or a dry sump is preferable. If you're committed to operation with a wet sump, then you're pretty much committed to the occurrence of steam explosion of some size. What size is yet to be assessed.

But the problem then arises of the possible endangerment of the engineered safety features within the containment. Again, it is not insoluble. It simply means that you do have to review how the materials are placed and protect them.

When it comes to backfitting, it simply is another cost to be addressed.

In other words, we're looking at this and it has all the earmarks of many of the other systems such as filtered venting, that have been looked at. It has its pluses and its minuses. We have considered it all along. I'm not aware that the French are considering it actively in their research program. It may be that EDF has looked more into it,but certainly, in the CEA research program, there is no indication of any work directed toward this type of device.

MR. SHEWMON: Well, there were several devices in the discussion, so I'm bothered some by it being talked about as a unique one.

To stay with that slide for a minute, Ivan Catton submitted to NRR a study about a year ago in which there was some talk about a rubble bed which was then floodable. Do you know whether that has ever been studied any further?

MR. KELBER: We're spending on the order of, I think it's a million and a half or two million dollars a year developing the dryout criteria and models for the mechanism of flooding of these beds. The work is being done at Brookhaven and at Sandia, primarily. We have supported --

MR. SHEWMON: These are sort of two-inch diameter balls?

MR. KELBER: No. We're looking at a range of diameters. Correlation has been established.

By the way, we also have cooperation with the Germans on this, who are doing out-of-pile studies. The correlation extends, so far as I know, on experimental data from very small particles of a millimeter or sub-millimeter size, through the fairly large particles that you mentioned.

MR. SHEWMON: Let me come back to what I think was Ivan't suggestion. That was that you take an inter-rubble, which would certainly not be put in that small a size, and that you then have the option of flooding so that you would have something to keep the molten fuel from coming in contact with the water immediately, but would also keep from closing off to get an insulating layer.

What it sounds like they were talking about is the coolability of rubble beds that are generating heat.

MR. KELBER: What he was talking about was simply a porous insulator.

MR.SHEWMON: But you would always have water underneath it,so you protect your core mat, your base mat, that way.

MR. KELBER: I don't know the extent to which this type of core catcher is being evaluated in any detail.

Essentially, this is a core catcher of an active type in which the porous insulator is used to protect the heat removal system. Those have been considered at various times, and I don't know the extent to which it's being considered now.

MR. CORRADINI: I don't know very much about it, just in conversation with Jim Fish at Sandia. I know they're doing large spheres with water saturation, which sounds very similar to what you're discussing. I don't know how far along those have gone, but I know some experiments have been done. They're not heat generated.

MR. SHEWMON: If we ever end up getting core catchers, I think that variety is much more desirable than what ended up in the flooded nuclear plant.

MR. DAVIS: This was looked at by EG&G with a mitigating circumstance with the Sequoyah plant. It looked like a reasonable approach. This study was sponsored by the NRC and came out about a year and a half ago.

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MR. KELBER: All of these concepts have their good and bad features. I think it is the type of study that the Risk Analysis Branch has carried out that has to be made to rank them. I think all of them have attractiveness, and I doubt very much that one can afford to make snap judgments.

MR. SHEWMON: How they are getting ranked is part of why we asked you here, I think.

MR. KELBER: I came here just to answer this question.

MR. CORRADINI: I'm trying to understand --

MR. KERR: The answer to the question is that we are studying methods for flooding containments?

MR. KELBER: Yes. The direct answer is that with regard to the French plans in this area, the only element of flooding or the only consideration of flooding in their national position is the flooding of basemat drains in three plants, not the 900 megawatt plants. What they will do for the larger plants, the N-4 plants, I do not know.

MR. CORRADINI: Just for understanding, is it only flooding that we are interested in or is it flooding and an adequate heat sink? You are automatically going to get flooding in a lot of cases just by having partial accumulation of ECCS.

MR. KERR: The person who raised this question is present.

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MR. SHEWMON: I am certainly in favor of heat sinks. What concerns me is the idea that once we have a severely damaged core, I have the impression that the safety studies sort of act as if everybody is sent home to have lunch to come back three days later to see whether it is melted through the core mat or blown the containment, and there are probably things that we could do and they would probably be easier if we would think about them ahead of time.

MR. CORRADINI: Let me suggest, in expansion of your question, to not only ask flooding but also maybe heat removal, auxiliary heat removal. As I was looking through here, I was just thinking, like TMLB', you are in a position where you are going to get some water in the cavity, and then when you have the failing, you have more water.

MR. KERR: I am going to suggest that if we are going to design this heat sink, we will let Charley sit down.

MR. KELBER: Let me answer this last one and then I hope I'm finished.

We are, of course, doing precisely the type of study that has been mentioned. Under the severe accident analysis program, or what is called SASA, we are producing a manual of guidelines for containment pressurization. I doubt that we can establish, for situations that will arise as rarely as this, operator guidelines of the sort that you

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have for the operational transients, but this manual will attempt to guide the operator into what systems and what procedures he can use and how long he has to effect repairs in order to deploy these systems to keep the containment intact, and at the present time we are not including the question of other engineered safety features which may or may not be required. We are addressing ourselves to the systems which are available.

Now, it may be that from these systems it will become apparent that there are additional systems which are available, but that may also come from other studies, and if other systems are very valuable in this regard, of course that will be just -- that was the type of information we are looking for. But this manual addresses itself to the question you have raised.

MR. SHEWMON: I'm not sure where the IDCOR people are going to come down, and I'm sure they would like to be able to convince people that the consequences and probability of the risk of this is so small that we need not worry about it or, on the other hand, do anything about it. If they cannot convince people of that, though, and they are concerned about station blackout sort of forcing them into multi-billion dollar backfitting arrangements, I would think they might look hard at what they could do to decrease the probability of station blackout. But that, then, was yesterday's

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discussion, and they probably aren't down to facing that one yet, from what I heard yesterday.

MR. KERR: Any other questions? Thank you, sir.

Did you and Mr. Shewmon have any further comments on the heat removal?

MR. CORRADINI: No.

MR. BASSETT: I see no reason why this particular segment could not be quite succinct, and depending on the extent of interest in the --

MR. KERR: You know, there are 32 ways that you can put a transparency on.

MR. BASSETT: I have already checked three of them.

All you have to do is use your common sense.

To summarize the purpose of this program, this is substantial, as you know. I have had occasion to discuss the length and breadth of it. Part of the start of it was three or four years ago when CORRAL was successfully attacked, MARCH-CORRAL was attacked by this same group, and at that time it became apparent that --

MR. KERR: Mr. Bassett, would you be willing to substitute constructive comment for attack?

MR. BASSETT: I think since the attack was well-founded and successful --

MR. BERNERO: Ruthless assault.

MR. BASSETT: At that point it became apparent that

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substantially better codes could be available and would be available as we got along with our experimental program.

I don't propose to read this slide to you or blab the results. I guess the important thing is that a lot of this work is pioneering. It takes time and it is quite expensive. We hope that it will come to a conclusion and afford us good and immediate results.

The technical issues which it addresses go to what happens if fission product release from the core, both in time and in chemical forms, and the mechanics involved in the aerosol formation and the attenuation mechanisms in the vessel and in the primary system, and the hydrogen release from the core -- the physical and chemical state of the core, including the progression of the melt, and what we can find out about coolability with reflood.

The program itself is -- this is a fairly busy slide but it is a summary of everything of major importance that we are doing.

We have integral in-pile tests at PBF and NRU. These are life reactor, real neutron tests and they give us actual rod bundle information. It is divided into two series at PBF: Phase one, which will be completed by April of '84, and phase two, which is now two tests, to be completed by June of '85, followed by the shutdown, as far as the NRC is concerned, of the PBF facility.

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We are currently contemplating two more tests to check the full-length test since it is the only place that we can get full-length fuel loads.

The separate effects program is in the ACRR and in Germany. Here we have an opportunity to watch a melt while it is in progress by optics, and the German are doing severe damage studies of fuel in the way of studies of the eutectics and various other phenomena that occur.

Also we are studying debris coolability in the ACRR reactor also.

This work is to generate a data base to support the development of SCDAP/MELPROG, and we believe it will be tied to TRAC.

Finally, the program consists of examination from selected samples of the TMI-2 core examination in cooperation with the DOE effort. It is not contemplated to be as grand as the DOE effort, but we are contemplating $2 million a year.

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MR. CORRADINI: The TRAC-MINUS program is new to me. Where does it fit in between SCDAP and MELPROG?

MR. MARINO: The TRAC code was developed by DOE, and it looks at TMI-type accidents. The early SCDAP code also does the same thing.

We're interested in linking feedback between the two and we compare it with a SCDAP and RELAP.

MR. KERR: M-i-l-a-s, that stands for million dollars or something or other?

MR. KELBER: Milas is actually one of the minor Greek gods and was the name chosen at Los Alamos when this work was originally started. It has no acronym to it.

MR. BASSETT: We had substantial foreign participation in the severe fuel damage program. We have U.K., the Netherlands, Italy,Belgium and the FRG. We are expecting cooperation with the British and we're looking forward to discussions with them in the next month or two. We have concluded our discussions with Japan. They're now on board with the program. They contemplate being with us for four years. We're still talking to Canada, Korea and Taiwan.

The total cost of the program is subsidized by the foreigners to the extent of about 15 percent in the years 1983 and 84,and we're also getting substantial in kind contributions in the metal area from the Germans, Netherlands and Belgium. So we estimate that the total foreign contribution to this program is about 25 percent of the total dollars.

We have had recently a significant successful test, the first scoping test, at PBF, and this is to indicate the sort of results that we are getting from these tests. The data is preliminary and is not verified, computer calculated and verified, in all ways. However, we can say that there was about 50 times greater rate of fission product release while liquefaction was underway than there was during the diffusion phase.

MR. KERR: Was that a surprise, or what was expected to be the case?

MR. BASSETT: We expected it to be the case qualitatively,but quantitatively it was somewhat of a surprise.

During quench there was, again, ten times larger release than what was happening during the earlier phase of liquefaction.

Iodine and telerium went in the liquid pathway and was recovered by that means. Iodine and cesium from this particular test --

MR. KERR: What is the liquid pathway from point to point?

MR. BASSETT: The point being that it was soluble, it came out in soluble form.

MR. MARINO: It goes through a separater tc separate the liquid phase from the solid phase.

MR. BASSETT: The iodine and cesium were recovered in the blowdown tank. The telerium was in the filter.

We didn't see any particular low volatility fission products at temperatures below 2400k. That is just about the point where the experiment is.

We found that SCDAP accurately predicted the thermal history and liquefaction of bundles during the test.

MARCH 1.1 did not do a good job. MARCH 2.0 got better results,about 200 to 400^{o}k.

The mass balance of the source term -- we won't really have a good handle on that until we finish the PIA, which is coming up in a few months.

MR. SHEWMON: There was some talk and, I think, definite plans to put silver cadmium alloys in here to simulate the control rods in most reactors or many. Did those get in this test, or is that a later version?

MR. BASSETT: Is a later version.

MR. SHEWMON: And this was test 1 or 2?

MR. BASSETT: This is what we call a scoping test to see how the systems work. We got quite a big batch of results.

MR. KERR: At the temperature to which you refer, at the 3800 and something Fahrenheit, that is below fuel melting temperature,you're getting some fuel dissolution and liquid zirconium. What fraction of the fuel entered into that reaction?

MR. BASSETT: I don't know. Do you know, George?

MR. MARINO: We haven't determined that exactly yet. The estimates are like 10 percent,something like that.

MR. BASSETT: In the pictures we can see some pellets, and we can also see some eutectic product.

The PIA will give a better indication,but you are doing neutron tomography,with some results which we haven't released publicly. We're still studying them.

The second study was due to go last week. We're still studying what we can do to fix some leak problems or run the test under existing conditions. As you know, these test trains are expensive,and we're anxious to proceed with a good bit of caution. We expect to have a decision as to our course of action within the next few days.

We have another test train in the pipeline. We'll probably get a test with the existing train,but I can't say that with certainty. We may have to go to the next test train.

The test which is now hanging fire and which we would like to get off as soon as we can will not quench; it will be in a steam atmosphere, and we hope to get the fission product release situation there.

The hydrogen evolution under these conditions -- these TMI -- so-called TMI heatup conditions we expect will increase the fuel liquefaction effect,and we will then go back again to look at SCDAP and MARCH and the MELPROG.

Just to indicate a littl[illegible]tail on the system -- and this is for the first [illegible] s at PBF -- this is the general system to separate gaseou[illegible] liquid samples and get the analysis that we gave you the preliminary results of earlier.

Finally, I would propose to indicate to you the changes that we have made in the program as a result of our determination of the funding available to us, which can be counted on over the years to come.

We're cutting down the degraded core coolability experiments at ACPR. We will be running three of them instead of the six that we had originally postulated. These experiments are very expensive and we think that three is a more cost-effective basis. It allows us to save some millions of dollars.

The NRC test matrix has been reduced from eight tests to two. We'd like to improve that. We think if we get foreign interest in this particular area, we can perhaps do a few more.

Finally, we're presenting culling the matrix for the second series of PBF which will amount to only two experiments where we had originally postulated as many as five, and our original plans at the start of this year were for three.

MR. SIESS: What does it cost you for testing ACRP?

MR. BASSETT: The one where we observed the bundle is one category. The coolability of the debris is another.

Bob Wright can give it to you.

MR. WRIGHT: The more expensive test, the coolability test, run about a half a million dollars apiece and they run for a couple of weeks,and they're substantial.

The degraded -- the debris formation relocation experiment, degraded core conditions, are very much simpler and they run at about $200,000.

MR. BASSETT: Here is the series 1, all of which we propose to run. We are, right now, quivering on the brink of running this one, and as you can see, we should have had it done by now,but we don't. And then there are three others.

MR. SIESS: The second column, is there any way of comparing the first test to the other four, since the units apparently are different?

MR. MARINO: The first test was the scoping test. We didn't intend to compare it. It has a much higher inflow rate than would ever be seen in these accidents.

MR. SIESS: But the heated rate is listed less than .5.

MR. BASSETT: It took about three hours to get --

MR. SIESS: Can you give me some idea where the heating rate designated TMI-2 is faster or slower?

MR. MARINO: It would be much faster. It would be maybe two degrees.

MR. SIESS: It would be much faster?

MR. BASSETT: Much faster than the scoping test, yes, sir. We're just feeling our way along. We had a few startling things happen, as it was, but we are proceeding very slowly.

This series, we think, will be completed. This series will not (indicating).

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We are certain to run series to number one, and we have not decided yet which of the other three will be the best one to run. Part of this will be determined by our analysis of the data from the first series. Also at this point the test force will be substantially down the road and there may be specific things we are looking for.

That is all I have to say.

MR. KERR: Are there questions? Mr. Shewmon.

MR. SHEWMON: Earlier there were -- in your summaries there were some references to the NRC experiments, and if you mentioned them later in the talk and what changes there were, I missed it. Can you enlighten me on that?

MR. BASSETT: We originally contemplated six to eight tests at NPU.

MR. SHEWMON: This is in the second round of experiments?

MR. BASSETT: No, it was sort of in parallel with the overall PBF. I would say more in parallel with the first round.

MR. SHEWMON: The first set on ballooning?

MR. MARINO: These were the severe fuel damage tests. We planned six. We have cut that back to two just to get confirmation.

MR. SHEWMON: And the customers are those in severe accident programs? The customer for the other was

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NRR. Whether they will use them or not?

MR. MARINO: Exactly.

MR. BASSETT: The argument about length is mildly persuasive but by no means of high priority.

MR. KERR: Further questions?

Let me ask one that is not related to your presentation except peripherally. I received a copy of a memorandum from Mr. Dircks to Commissioner Asselstine under the general heading of damage to core cooling, and in response to Mr. Asselstine's question as to the Staff's current views on why during the early periods of the TMI-2 accident, the core did not degrade substantially more.

In answering that, the statement is made that at core temperatures in excess of 3600 degrees F., among other things the molten zircaloy will dissolve a significant fraction of the UO_2, and then it says such damage was predicted by the Fuel Behavior Branch a few weeks after the incident and subsequently by other studies, and was later confirmed when the TMI-2 core was examined by a small video camera.

I thought when we saw those pictures and heard comments on them that the statement was made that there no fuel melting.

Apparently I misunderstood or later the results --

MR. MARINO: The statement was made, I think, by

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the NRC Staff that the fuel did not reach the stoichiometric UO_2 melting point, which is 5000 degrees Fahrenheit. The statement was made that some of the fuel liquefied in the form of interacting with the liquid zircaloy and they had a crest on the top of the TMI core which was very difficult to penetrate. It is not loose melting.

MR. KERR: When I saw the picture, there was some discussion, some small pellets that were said to be zircaloy, but I thought the statement was made that we didn't see any melting.

MR. BASSETT: I think that is semantics.

MR. MARINO: Melting of cladding occurred and dissolved the fuel.

MR. KERR: But this says that there was fuel melting in a sense. If the fuel was dissolved -- fuel, if you run steam against sugar, you will liquefy it below the melting point of sugar, and I think you are getting into that sort of a phenomenon. You know, they using liquefying in here, not melting. So as you change composition, you can lower the melting point.

MR. KERR: It liquefied but it didn't melt. I feel better.

MR. MARINO: If someone says UO_2 melts, it will melt at 5000 degrees Fahrenheit. If you have liquid zircaloy in its presence, the liquid zircaloy will dissolve some of

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the UO_2.

MR. KERR: It was observed here. There was some indication that the temperature reached 3800 or so.

MR. MARINO: If you will look at the pass-out we gave you on the neutron radiograph, the scoping test, you will see a ball of what we think was liquefied fuel and cladding.

MR. KERR: They probably had some independent measurements of temperature, or did you?

MR. MARINO: Yes.

MR. KERR: For TMI-2 you didn't have any independent measurement of temperature?

MR. MARINO: Exactly. We had to calculate temperatures based on what we thought the coolant level was.

MR. BASSETT: We had a measurement of up to 2200 and we had a calculation before that and extrapolated to the point where we scrambled.

MR. KERR: Mr. Lee.

MR. LEE: Do you have, by any chance, some kind of list of items of models that you anticipate would be verified by the end of phase one, PBF tests, for example? The model uses the TRAC code or anything like that?

MR. BASSETT: I don't know that we have on one piece of paper covering all of the assessment and validation programs for these codes. We could prepare such if it was

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useful to the Committee.

MR. KERR: Would it be two or three pieces of paper?

MR. BASSETT: That is why I volunteered to prepare it. We can certainly put it together in a hurry.

MR. KERR: It would be helpful if it is not a lot of effort.

MR. BASSETT: We can do it.

MR. SHEWMON: If I might introduce an interesting diversion here, are we anyplace closer to seeing more of the TMI-1 core, or is that scheduled for -- pardon me. I don't want to start any rumors. Is that going to be in this calendar year?

MR. BASSETT: It's the next, isn't it?

MR. KERR: Is there someone here who can respond to that question?

MR. KELBER: The DOE have proposed to us a matrix of tests and their associated costs and priorities. The Staff is preparing for examination by the office directors and Mr. Dircks a set of alternatives to select from that for NRC funding far away from the site examinations.

When it will start may well depend upon the funds available to both agencies. My guess is that you will not see any samples pulled this calendar year. That is my best guess.

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MR. MARINO: I would like to that the major contractor for the core examinations are the EG&G staff, who are doing severe fuel damage tests in the PBF program, so they will be able to correlate the known temperatures of the PBF tests to what they see in the examination of the TMI core. We have got to go into that core and we will not have a good idea of what really happened to it, and if we have some well-characterized data behind it, some other tests and a common contractor is doing it, I think we will get a lot of information.

MR. KERR: Other questions of Mr. Bassett or otherwise?

Thank you, gentlemen.

We have a lunch recess scheduled at this point, and we come back --

MR. BERNERO: Dr. Kerr, I was wondering if you could go into where do we go from here on severe accident. I think I told you separately that we have extensive comments from NRR on NUREG-0900, and as soon as we have the fiscal 1985 budget decision in hand, we expect to prepare revisions under NUREG-0900. That would be toward the end of this year.

END of ft/ar10

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I would like to know, if I could, where do you foresee the Subcommittee's interest lying between now and then?

MR. KERR: It is a legitimate question. I don't think I could give you a good answer at this point, but I will try to give you one shortly.

Any other questions?

We will recess until -- what do we have? Mr. Fleischman is scheduled for 1:45. We will recess until 1:45, at which time we will hear a discussion of hydrogen control.

(Whereupon, at 12:40 p.m. the meeting was recessed, to reconvene at 1:45 p.m. the same day.)

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AFTERNOON SESSION

(1:45 p.m.)

MR. KERR: The meeting is in session.

MR. FLEISCHMAN: I am here to talk to you today about the final, final rule on hydrogen control, hopefully. I was here about a year and a half ago to talk to you about when we were going out with it as a proposed rule, and also when we were talking about another rule that was going out effective last December.

Just to give you a little basis --

MR. SIESS: Could you help us out just a little bit? We have got a copy of the draft SECY, I guess, marked April 22. We were just handed one marked April 25. Are there any substantive changes in them? I have already read and marked one of them up.

MR. FLEISCHMAN: There are no substantive changes. There are just some little word engineering things, some comments from the lawyers, stuff like that. In fact, there probably haven't been any real substantive changes since February. Everybody likes to noodle it a little bit to make it a little better.

MR. SIESS: I will save the original one.

MR. FLEISCHMAN: Just to bring you a little up to date, a reminder of what we have done in the past. There have been several rulemakings that have been issued recently

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in the past couple of years related to hydrogen control, and there was the advance notice for long-term rulemaking, which was October 2, 1980. And at that time we also had a proposed rule that was issued on October 2, 1980. That proposed rule covered many, many separate items related to accident monitoring, including hydrogen control.

At the request of the Commission, we were told to limit that rule to only hydrogen control items, and the final rule that was issued on December 2, 1981 was limited only to inerting of Mark-1's and 2's, hydrogen recombiner capability, and high point vents.

We have previously discussed with you the proposed rule on hydrogen control that would apply to Mark-3's and ice condensers. There would be equipment survivability requirements for all plants in which burning was a possibility, and also we were going to require analyses.

Basically that proposed hydrogen control rule was to formalize regulatory decisions that were already being implemented in licensing actions such as for Sequoyah and McGuire.

Is that legible? There was a policy statement on severe accidents that was just recently issued, and what we are here for now is to discuss this final rule on hydrogen control and equipment survivability qualification. The one that was proposed on December 23rd. We are here to discuss

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what we are to do with that now for the final phase.

MR. KERR: December 23 of '82?

MR. FLEISCHMAN: December 23 of '81.

MR. SIESS: Don't get them moving too fast.

MR. FLEISCHMAN: Just briefly to show you this, the first hydrogen control rule, again, was involved with the inerting of Mark-1's and 2's, external hydrogen recombiner capability, and high point vents. That is the one that was issued on December 2, 1981. That is effective -- let's see. That was effective May 4, 1982 for the inerting, so all Mark-1's and 2's are inerting.

This present rule is going to be considering just Mark 3's and ice condensers.

MR. SIESS: There was only one Mark-1 that that applied to?

MR. FLEISCHMAN: Two. Vermont Yankee and Hatch-2 were the only ones that were affected.

As far as this proposed rule is concerned, the comment period was extended an extra 60 days to April 9, 1982. We have had 28 persons submitting comments, about 202 separate comments, and the detailed comments are in enclosures of the Commission paper which you have, and we are in the process of trying to get the final rule revised now as a result of the public comments that we have received.

There were a number of comments, like I say, that

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are summarized in your enclosure there. I would say the major comments were to the further rule, to the severe accident rulemaking when research that is being worked on now and PRAs are completed. People had problems with the 75 percent metal-water reaction. They thought it was too large. People had problems with the two-step approach to equipment qualifications. There was a question about whether or not the equipment needed for safe shutdown should be needed for safe cold shutdown.

There was that question, and there was also a problem with the implementation schedules. They felt that they were unrealistic. Those were the only major changes -- or the major aspects that we considered in revising the rule.

MR. LEE: Could you elaborate a little bit on the circumstances surrounding cold shutdown versus safe shutdown?

MR. FLEISCHMAN: There is different equipment that is necessary for cold shutdown versus safe shutdown, and there is actually work being done now -- there is an unresolved safety issue. I think it is Task A48 or A45 -- A45, that is looking into that. Essentially we felt we didn't want to get into that in this rule, that we would defer any question of safe shutdown versus safe cold shutdown to that unresolved safety issue.

MR. LEE: Thank you.

MR. SHEWMON: On the question of inerting, did there

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ever get to be anything like a cost-benefit analysis on this or was it just an administrative decision that it would be done, an engineering judgment I guess that usually gets called.

MR. FLEISCHMAN: I guess it was primarily engineering judgment. I don't think there was any cost-benefit analysis done on inerting.

MR. SHEWMON: Utilities were arguing that if it wasn't inerted, they could go back in and could do maintenance and check on things that they normally would and it would help plant safety, in a sense. Of course, the odd chance of somebody getting in the wrong place --

MR. SIESS: Those arguments were never well-coordinated by the utilities, and when Vermont Yankee was in and somebody asked them how often they went in, it turned out it wasn't very often. Of course, the people that were inerted couldn't contradict that because they didn't know how many times they would go in.

MR. SHEWMON: Were people who were already inerted -- had they known they were going to do that before the plant -- so that they could put certain things outside of that inert atmosphere that are now inside for the others?

MR. SIESS: All the BWR-1's and 2's -- the 1's were all inerted from the beginning.

MR. FLEISCHMAN: My impression is that the people joy7

that were inerted, the Mark-1's and 2's, they haven't really tried to request that they be de-inerted now. In fact, because of the requirement for equipment to function during the burn, I think the fact that they were inerted, the rule doesn't apply to them. So I think a lot of them are taking advantage of the fact that they are inerted now.

MR. SIESS: I don't recall ever having heard a discussion as in the arguments for requiring recombiners for the plants that were inerted.

I don't want you to get into a long discussion of it, but is there a good reference you can give me on that or provide me?

MR. FLEISCHMAN: It is my understanding that that is still under discussion and Staff is considering it.

MR. SIESS: That is not a part of the rule?

MR. FLEISCHMAN: It is not part of this rule. It was part of the final rule that was issued on December 2, 1981. At the time that rule was issued, the Mark 1's and 2's were -- that rule required that all plants that relied on a purge repressurization system as a primary means of controlling combustible gases have its capability --

MR. SIESS: That is the CAD system, isn't it?

MR. FLEISCHMAN: Yes.

MR. SIESS: That is not inerted. But there was a rule proposed that went to Mark 1's and Mark 2's that were inerted that went to recombiners?

MR. FLEISCHMAN: This rule was interpreted by the Staff, and the intention of the Staff was that Mark 1's and 2's would have to have the capability to install an external recombiner following an accident. Many of the Mark 1 owners complained about that, and they have asked for an exemption from that aspect of the rule. It is my understanding that the Staff is still looking at that.

Could you say anything more about that, Charley?

MR. SIESS: If it is not in this --

MR. FLEISCHMAN: It is not part of this rule. It may very well be a good subject for a future meeting.

MR. SIESS: I thought the containment atmosphere pollution was a PWR solution.

MR. TINKLER: There was some confusion as to plants that rely on pressurization as a primary means of hydrogen control. What that phrase was intended to mean was the purge repressurization system, which was the sole active system to control hydrogen for a plant which was inerted, but for long-term hydrogen control relied upon purge repressurization. That was still included in that category even though the plant was inerted.

MR. SIESS: I don't know where the confusion was. It is not in the present rule, so let's don't take time on that.

MR. FLEISCHMAN: Instead of showing you what the proposed rules are, I thought I would show you what the rules are right now and what changes are now. The final rule the way we have it now would have hydrogen control only for Mark-3's, and ice condensers, and it would be effective two years after the effective date of the rule rather than one year. We originally were talking about implementation for the analysis and the actual implementation of the changes within one year. Now what we are suggesting is that the analysis be done within one year and the actual implementation be done within two years.

So as you can see, the first change was changing the 1 to a 2, and that was also the main change as far as implementation on that part is concerned.

As far as the equipment survivability question -- and we are really talking now about the survivability of systems that function -- systems and components that have to function during or following a hydrogen burn. That is what we are really talking about there.

Originally that was also supposed to be one year after the effective date of the rule for the ice condensers. We changed that to two years. That also was to apply to all PWRs, all light-water reactors, and it was going to be effective for them two years after the effective date of the rule.

The change we are proposing now is that all LWRs other than Mark 3's and ice condensers be excluded from the rule. The rule would only apply to Mark-3's and ice condensers. Ccher light-water reactors would be considered in the long-term rulemaking on severe accidents. The feeling was that the higher pressure capability and larger volume of the large drives -- they could only handle the hydrogen generator from a degraded core accident without having a problem.

So the rule is going to only apply now to Mark-3's and ice condensers.

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You're satisfied,but you're going to continue to look at that as a part of the severe accident.

MR. FLEISCHMAN: Right.

MR. KERR: Is that based on the assumption that the hydrogen burn is not a problem or that you wouldn't get hydrogen burn in large drys?

MR. FLEISCHMAN: I think it's based on the assumption that a hydrogen burn, even if you had it in a large dry would not be a problem. Based on the improvements that have been made to the large dry since the Three Mile Island accident, and the fact that the Three Mile Island accident equipment that was necessary to shut down did seem to function during and following the burn.

MR. SIESS: Inherent is that is that equipment that's qualified for the LOCA has a good chance of surviving a hydrogen burn. Is that part of it?

MR. FLEISCHMAN: That's right.

MR. SIESS: Assuming that the equipment is qualified for a LOCA, all of the equipment is in the process of being qualified but not all of the equipment has been qualified.

MR. FLEISCHMAN: What we're saying in the survivability concept or the qualification concept that we're talking about -- and we discuss it in detail in the paper -- is that equipment -- it has been shown so far that equipment that has been qualified for a LOCA has had a thermal response during hydrogen burn which has been enveloped by the thermal response that they found for the qualification test. So therefore,the feeling was that it would also survive a burn.

MR. SIESS: The emphasis there is on thermal response rather than ambient temperature?

MR. FLEISCHMAN: Yes.

MR. SIESS. The shorter duration?

MR. FLEISCHMAN: Right.

MR. DAVIS: Excuse me. I have a question on the current rule.

MR. FLEISCHMAN: The current rule?

MR. DAVIS: The final rule. Correct me if I'm wrong,but it seems to me that if you don't inert, then you have to provide equipment survivability for an H-2 burn.

MR. FLEISCHMAN: Correct. Except if you are a large dry, a PWR with a large dry.

MR. DAVIS: What if you determine that there's another way to prevent burn besides inerting. Do you still have to guarantee equipment survivability for a burn?

MR. FLEISCHMAN: The way the rule reads, as long as you don't rely on inerting the containment,then we're saying that the way the rule reads it would be that we're assuming that it is going to be a burn. It doesn't have to be pre-accident inerting. It could be post-accident inerting, as well.

In other words,they can come up with a system that once you have the accident,you inert the containment at that time. That would also mean that you didn't have to worry about a burn.

MR. DAVIS: The rule does not allow any other means to prevent burn other than inerting. It would be senseless to do so because you still have to qualify your equipment.

MR. FLEISCHMAN: The way the rule is written, that's correct.

MR. DAVIS: There are other ways to prevent burn besides inerting. It seems like you've excluded those possibilities with this rule. That was the intent, I guess.

MR. ROSZTOSCZY: The intent of the rule would be that you require qualification of the equipment if you are not preventing hydrogen burn. If you have a system that prevents hydrogen burn, then you do not require qualification. The only thing that I'm not clear on in the answer was apparently that right now, there's nothing asked in the rule with inerting --

MR. DAVIS: The rule clearly states in Item B, page 2, that if you don't inert you must assure equipment survivability. It doesn't say if you don't have hydrogen burn.

In other words,the only mechanism that's allowed in this rule for prevention of burn is inerting.

MR. ROSZTOSCZY: I'd say it a little differently and say that the only mechanism accepted at the present time is inerting, and what it means,then, if somebody comes forth with a different way of preventing hydrogen burn and it's accepted, then the rule will have to be modified at that time.

MR. DAVIS: Right now, it could not be accepted under this rule.

MR. ROSZTOSCZY: Right. So the rule needs to be modified.

MR. SIESS: Right now, the only means you would expect is inerting.

MR. FLEISCHMAN: That's correct.

MR. SIESS: If somebody comes up with something else and you end up accepting it, you will change the rule?

MR. FLEISCHMAN: That's exactly right.

MR. DAVIS: Why couldn't the rule say inerting or an approved means of preventing combustion?

MR. SIESS: The industry would probably tell them what is an approved means and get it in the rule so they can get on with their businss.

MR.DAVIS: This way, it would be approved already, or anything equivalent. It seems like you're closing out options.

MR. FLEISCHMAN: I see your point. It seems like it was a simple way of doing it. So far,the only means that have really been seriously proposed has been inerting. So we thought it would just simplify the thing.

There's a halon suppression system that has been looked at. There have been other systems,but it was felt to be more direct to just make it, say, inerting.

Now, the intent, of course, if you came up with a system that proved to us that there would be no burning, then you wouldn't have to qualify the equipment.

MR. DAVIS: What about external recombiners? Would they be excluded, then, under the present rule?

MR. FLEISCHMAN: Yes.

MR. SIESS: As the sole means?

MR. FLEISCHMAN: Right.

MR. SIESS: They are not excluded; you can use them.

MR. FLEISCHMAN: If you had an external recombiner, that wouldn't be grounds to avoid having to show that your equipment can survive a hydrogen burn.

MR. KERR: Are you talking about an external recombiner that would handle a 75 percent metal-water reaction that occurred over a brief period?

MR. FLEISCHMAN: As far as I know,the recombiners don't handle that.

MR. KERR: Most recombiners don't come close to that.

MR. DAVIS: But you could postulate one that could. I've been involved in designs of recombiners,and you can make them that big. There are also people working on hydrogen getters that would remove the hydrogen from the atmosphere without burning it.

MR. KERR: You're not talking about off-the-shelf equipment; you're talking about a development program, or are you?

MR. DAVIS: No,it wouldn't be off-the-shelf equipment.

MR. SIESS: I think we have to make a distinction. The way you're thinking and the way I think is a little bit different. You say it wouldn't be allowed under the rule, and that is correct. I think the staff, and to some extent the industry, tend to think more of what would be accepted under the rule rather than what isn't allowed under the rule, and the emphasis is on the positive. If you do this, it's okay.

Now, if you want to try something else, you can come in and argue it.

MR. DAVIS: I remember Appendix K experience. It seems to me like we're getting into the same problem here. This is acceptable, so you really force everybody to go this way. You don't allow anything else to be acceptable unless you change the rule. As I understand it, that would take some time and quite a bit of work if you wanted to make an exception.

MR. FLEISCHMAN: The MARK-3 is an ice condenser, apparently. They have decided that the hydrogen control system they wanted to use has been an ignition system, so they're all going to try to meet the rule this way. They haven't gone to an inerting system.

I think part of the reason we worded it this way was we said look, if you wanted to go and inert your containment, you could do that. But so far,they have all decided to go with the distributed igniter system.

VOICE: Let me comment on that in terms of the time that it might take to change the rule.

There's also another approach available for that case. They may ask for an exemption. Any applicant can ask for an exemption and can go forward right away based on the exemption, if it is granted.

MR. DAVIS: Thank you.

MR. FLEISCHMAN: A little more discussion of the actual requirements of the rule and the changes. The rule will require hydrogen control systems for MARK 3s in ice condenser plants with a 75 percent fuel cladding-water reaction, with no loss of containment integrity. Previously,we said it had to meet the ASME service level C or factor load category limits.

We have modified that in the rule now to say that it should be done by an accepted method; a method that had been accepted by the NRC Staff. And, for example, they could use actual material properties with margins and do a more realistic calculation, or they could use any other method that they could convince us is reasonable. So we are allowing an option of other methods besides just meeting the ASME service level C limits.

The rule originally applied also, as far as the functioning of systems and components during a hydrogen burn -- it applied to all non-inerted LWRs. Now we're limiting the rule only to MARK 3s and ice condensers.

The other thing was we were saying this would be for systems and components needed to establish a safe, cold shutdown. We modified that to say only safe shutdown.

Furthermore, the original rule required that local detonations be included. We have modified that to say that if they could show that local detonations are unlikely to occur, that they don't have to consider local detonations in their analysis.

The other major change is the analysis that was required for the rule before it was to justify the selection of the hydrogen control systems, and we were looking for a comparative analysis of alternative hydrogen control systems. We have modified the rule and the response to any comments so that the analysis that would be required is just the analysis to actually support the selection -- to support the hydrogen control system selected, just to show that the one selected was adequate.

MR. KERR: Let me see if I understand. Is that expected to be a different analysis for almost identical plants? Does each plant require a plant-specific analysis, or is the analysis selected to be rather generic for similar plants?

MR. FLEISCHMAN: I would say for similar plants, they could rely on generic analyses.

Do you have any comments on that,Charlie?

They would have to show that their planning agreed --

MR. KERR: But if plant A has an A-type budget and plant B has a B-type budget, do they have to do different analyses, or if you have a large dry that is just like another large dry -- I'm trying to get some feel for the tail of analyses that you expect.

MR. TINKLER: We would expect plants that are very similar to rely upon generic analysis.

MR. KERR: Thank you.

MR. LEE: Could you elaborate a little bit more on the type of analysis that would be required in light of the various computer models that are being developed for containment analysis and things like that?

MR. FLEISHMAN: You mean the details of the analysis?

MR. LEE: Right. Would some codes that are in use now be acceptable?

MR. FLEISHMAN: I would say they would be. They are doing analyses right now on these plants, and they have been doing analyses on the Sequoyah and McGuire plant, and that sort of analysis has been acceptable and would be acceptable.

MR. CORRADINI: So it would be a plant-by-plant checkout of their methods in meeting the rule? Every plant com ing up could conceivably have a different method of analysis?

MR. FLEISHMAN: They could if they wanted to have a jag89

different analysis. They could discuss it with the Staff and if the Staff approved it, they could use that analysis.

MR. LEE: Could you also comment a little bit on those two tables you have included in the proposed rule? How you arrived at those tables, or how the licensees are expected to utilize those tables?

MR. FLEISHMAN: The tables that were -- they are not in the rule, by the way. They are in the statement of consideration, the preamble. It is not part of the rule. It is a suggestion. It is the same as a regulatory guide, you might say, and those tables are suggestions. They are arrived at by, I would say, a concensus at a meeting within the Staff of those -- those are the scenarios which we felt were the most significant as far as the probability of generating large amounts of hydrogen.

MR. KERR: Which table is that?

MR. FLEISHMAN: Between page 15 and 16 in the enclosure.

MR. LEE: The latest one is the proposed rule of February 1982, Federal Register.

MR. FLEISHMAN: It's also in your report.

MR. LEE: Right. The body of the report, too.

MR. KERR: Table 1.

MR. LEE: Table 1 and Table 2, both of them. Table 1 is not on the Federal Register that I quoted.

MR. FLEISHMAN: I think it was. They were both in the Federal Register.

MR. LEE: But Table 2 is quite specific for pressurized water reactor, but you don't have anything equivalent to that for boiling water reactor. I'm trying to understand what is the rationale for picking up this particular table for a PWR.

MR. FLEISHMAN: It was a suggestion. This sort of variation had been used by people who had been doing the analysis of the ice condenser. We didn't have a suggestion for the PWRs.

Do you have any comment on that, John?

MR. LONG: That is exactly right. The PWR owners disagreed among themselves to some extent, at least initially, as to what procedure they felt would best take advantage of the situation that they had at their particular plant, and they wanted to have the option of using different procedures, and we tried to accommodate them by allowing this choice. The PWR owners seemed to accept the idea that they would select a group of scenarios that we could live with, and so far it hasn't been necessary to exercise that choice with PWRs. If it becomes necessary, then again we would have to consider that, and it might lead to a further modification.

MR. CORRADINI: So this is a guideline table that they don't have to necessarily follow, and it is only for PWRs?

MR. FLEISHMAN: That's right. There is no -- no Table 2 has been submitted for the PWRs.

MR. CORRADINI: So it is up to the owners' groups, or individual utilities to come up with their own release rates?

MR. FLEISHMAN: That's right. And the Staff would review it. In fact, if you review the text, we don't recommend any particular method. We are suggesting that they could use either the method of Table 1 or the method of Table 2, and we are suggesting scenarios. If they feel for their specific plant that they have some other accident scenarios that would be more likely to produce hydrogen, they should use those scenarios.

MR. SIESS: In the proposed rule, on page 22, paragraph 6, subsection B, it says the analysis required by paragraph, et cetera, must -- and then I go down to item 3, and it says "must use accident scenarios."

That is all the guidance the rule is going to give.

MR. FLEISHMAN: Where were you?

MR. SIESS: Page 22. It's a few lines up from the bottom.

MR. FLEISHMAN: Must use accident scenarios.

MR. SIESS: That's all. The next item is 4.

MR. FLEISHMAN: I wonder if there is a typo there.

MR. SIESS: I can't make any sense out of it by putting words back in, either.

MR. FLEISHMAN: Maybe something is left out.

I think there has been something left out of there.

MR. SIESS: I looked at the April 25th one and it's the same thing.

MR. FLEISHMAN: It probably should read "accident scenarios that have been accepted by the NRC Staff."

MR. SIESS: Acceptable to the NRC Staff maybe shouldn't have been deleted.

MR. FLEISHMAN: We've changed it to "that have been accepted by the NRC Staff." I'll check into that. I think that is just a typo. That is what we meant, though.

MR. SIESS: That's what you meant. All right.

MR. FLEISHMAN: Right.

MR. SIESS: I'm sure one of the Commissioners will note that.

MR. FLEISHMAN: That looks like a typo that was left out. I'll check into that.

MR. CORRADINI: Could I ask a question about something else?

I'm just kind of curious. I think I understand all the changes except local detonations included unless unlikely to occur.

What is a local detonation? That is, versus -- is it essentially in a small region of the containment versus the global?

MR. FLEISHMAN: Yes.

MR. CORRADINI: And then unless unlikely to occur. Before that was put in, then local detonations of some size were going to have to be considered by the rule?

MR. FLEISHMAN: Unless unlikely to occur. What we had before, they would have been forced to assume that there was a local detonation. No matter what they might have done, what design features they may have had to prevent local detonations, they would have been forced to include local detonations in their analysis.

MR. CORRADINI: So the rationale in changing it was that now there is physical evidence of local detonations that cannot occur in some situations?

MR. FLEISHMAN: That's right. There are design features that you can install to prevent local detonations occurring. Fans and things like that. And certainly to prevent a buildup of hydrogen concentrations.

MR. LEE: If I may come back to Table 2 again.

Do you feel that it was justified to put this much detail in, even as a guideline, as a part of the proposed rule?

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MR. FLEISHMAN: It is purely a guidance to them.

MR. LEE: But it will be part of the Code of Federal Regulations?

MR. FLEISHMAN: It is in a separate part -- it is actually not part of the regulation itself. This is guidance that goes along with it. The rule itself doesn't start until you get to page 17, I guess. Actually, it doesn't start until you get to page 18. That is in the Code of Federal Regulations. This is a preamble and would show them how we interpret the rule. It is giving the licensing guidance on how to meet the rule.

MR. LEE: So it is inthe Federal Register but not in the Code of Federal Regulations?

MR. FLEISHMAN: That's correct.

MR. KERR: Do these analyses give a rate of release? Is that what the analysis is for,to show that the mitigation system can deal with it?

MR. FLEISHMAN: To show that the hydrogen control system can handle the hydrogen.

MR. KERR: Suppose I find out that this gives me an 80 percent water reaction? Then what do I do?

MR. FLEISHMAN: What we are doing is, we are giving rates of release. We give rates of release in column 1. We give the rate of release in column 1. We are saying you would take that rate of release until you ended up with a

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75 percent metal/water reaction and then cut it off.

MR. KERR: So you don't really have to do any --

MR. FLEISHMAN: You don't have to analyze the accident scenario in Table 1 if you decide to use the procedure of Table 2. If you decide to go with Table 1, then you would have to calculate what your actual hydrogen release rates would be.

MR. CORRADINI: Could I ask another question, then? Then maybe I don't understand. This final rule is something still interim in relation to what is going to happen with severe accident rule-making, or decision-making, or is this essentially it for Mark IIIs and ice condensers in terms of hydrogen control?

MR. FLEISHMAN: I would say this is it for ice condensers and hydrogen control. It is the same as any rule published. Any rule can be rescinded at a later date. In fact what we are doing here is amending regulations that already exist. So at the time of the severe accident rule-making, if we found for one reason or another that we wanted to change it, we would change it. I don't think we are anticipating making any further changes in it right now.

That is essentially it. This last one sort of summarizes the changes which we have made which we have seen on the previous two Vu-Graphs. We've essentially deferred the

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requirements on the dry containments until the severe accident decision. We've revised the implementation schedule to make it one year for the analysis and two years for the implementation.

We are allowing other acceptable methods for showing containment integrity besides the ASME code. We are eliminating the requirement for cold shutdown.

MR. SIESS: You say you are allowing something other than ASME code.

MR. FLEISHMAN: We are allowing them to use another method to show that containment structural integrity will be maintained other than meeting service level C factored load category. We are allowing them to use some other method that they may want to propose that we would consider acceptable.

MR. SIESS: I'm sorry. I still read 4(b)1, steel containments meet the requirements of ASME boiler pressure vessel code, incorporated, et cetera. This is page 19, paragraph 4, subparagraph A, subitem 1.

MR. FLEISHMAN: Look at B first.

MR. SIESS: That talks about a method.

MR. FLEISHMAN: This method could include the use of actual materials, properties, with suitable margins, and so on.

Another method could include a showing that the

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following specific criteria of the ASME boiler and pressure vessel code. We are using that as an example. That would be one method they could use, or they could use some other method.

MR. SIESS: The boiler code covers both steel and concrete, right?

MR. FLEISHMAN: Yes.

MR. LEE: Was any specific model suggested in lieu of the ASME code service level C that instigated this original clause?

MR. FLEISHMAN: Yes. There has been analysis done, I think, for the Sequoyah and McGuire plants, in which I think they actually did actual calculations. In fact, the Staff has done actual calculations using actual material properties, and used statistical combinations of various methods to show that the containment would survive with margin.

Do you want to say anything more about that, Charlie?

MR. TINKLER: I think you characterized it.

MR. LEE: How does the analysis compare with the analysis performed according to the ASME code?

MR. SIESS: The ASME code doesn't have analyses for strength. They only have analyses for design. What they did was make an actual analysis to determine when it would fail.

MR. LEE: Right. But to resolve whether it should satisfy the ASME code?

MR. SIESS: ASME is a design code, not a failure calculation code.

MR. LEE: What if they come out with either stress or strain or whatever could be compared with the ASME code?

MR. SIESS: You could probably determine at what service level it compared to, whether it was C or C and a half, or D or D and a half, or something like that. They attempted to make a failure analysis. This is a little hard to do with a design criteria.

MR. LEE: So service level D could be acceptable, providing you perform the analyses and can somehow show that the structure will survive?

MR. FLEISHMAN: I would say if they could show that the structure could survive and present risk data to show that it would survive with certain probability and convince the Staff that that was satisfactory, and they only met service level D, that that would be okay, also.

MR. LEE: Thank you.

MR. CORRADINI: I have a question. This is more overall. I'm reading some of the utility comments that I've gotten before, and I'm looking at SECY 82-1-B. I guess I take some of the utility comments, I take to heart, on the one hand. They see a dichotomy. On one hand, this is a specific rule for a specific thing, hydrogen control.

On the other hand, they make the point that under the severe accident and rulemaking and under the policy drafts which I guess they have seen, probablistic approach to an accident of this magnitude of hydrogen release should also be taken into account as well as the release itself.

So, I'm a little confused. This rule essentially postulates a source term for hydrogen and then worry about how the containment would be threatened, and some of the comments suggest that this may not be one of the dominant sequences. Getting to this poing during the degraded core accident may not be a dominant sequence.

MR. FLEISCHMAN: May not be a large risk contributor.

MR. CORRADINI: On the one hand, I read the proposed Commission policy statement, which I gather was drafted by the regulatory group somewhere in there, and it seems to be different in spirit than what this is, which is essentially a prescription based on engineering judgment, but a prescription on how to attack a certain level of physical processes which are beyond the design basis. Not probabilistic. So an interim rule that is totally different than what is being now put out as policy. So I am confused.

MR. FLEISCHMAN: I don't think there is a difference, really. I think here we say that we have had a problem, we have had a Three Mile Island accident. We know that we have generated a 45 to 50 percent metal/water reaction and we say we want to make sure that even though that is supposedly an unlucky accident, the Commission, I know feels that way, and I think the Staff does also, we feel that we should make sure that we have means to protect against that sort of an event.

So, whether or not it may be a low probability or not has been shown to occur, and we feel that we should have a means to mitigate the effects of such an accident.

MR. KERR: I'm puzzled here. Is there a difference between something being a low probability and having shown to occur?

MR. FLEISCHMAN: You might say maybe we don't have that much faith in the probability analysis; I don't know.

MR. KERR: I have suspected that for years. But, you know, the fact that things are low probability doesn't mean that they won't occur.

MR. ROSZTOCZY: A few comments along the line that you are talking about. The --

MR. CORRADINI: I'm really asking the question just out of ignorance more than criticism. If I were a utility in the Midwest or somewhere and I read this, and then I see the policy statement where PRA is being pushed, or recommended as a tool of judgment, then I would tend to be a little bit confused. And I read the comments by the utility people here, the hydrogen control owners' group, and I sense that just by reading what they are saying.

MR. KERR: The problem is you are looking for consistency. Many people have said consistency is a refuge for small minds. I don't know to whom you attribute it. This is not consistent, but deemed to be good engineering judgment, even so. It happened, and I guess it could happen again.

It is as simple as that.

MR. ROSZTOCZY: All of our determinations are based on a deterministic approach at the present time. As you know, it is going to stay that way for a minimum of two more years. This rule is a rule that will become effective in the near future. This is simply an extension of the existing deterministic approach and the existing deterministic rules. Separate from this issue, we are examining whether there is a need to have something more beyond this, and if there is a need, then what shape or form should it take, including probablistic together with other possibilities.

MR. KERR: You have to recognize that Mr. Rosztoczy is a part of a different organization than one that requires a certain reliability level for auxiliary feedwater systems. That particular one is reliability based. But that is a different organization.

MR. CORRADINI: I understand. Thank you.

MR. SHEWMON: You are not sure you believe, but you understand.

MR. CORRADINI: I'm trying to get the origin of SECY 82-1-B, and I gathered it was from the regulatory group, maybe a different part of the regulatory group.

MR. KERR: Well, you find 82-1-B, especially when it deals with existing plants, to be liberally laced with engineering judgment, and the approach, as I am beginning to understand it from Mr. Bernero's presentation this morning, is that one does PRAs and one looks at experimental data, and anything else one can get, and then one uses engineering judgment.

Now, whether this is engineering judgment based on PRAs --

MR. CORRADINI: Or just the fact that it has happened.

MR. KERR: It's difficult to determine. In some senses, it is inevitable, I suppose, given the current state of the PRA business.

Would you help me a bit? I'm trying to recall what the present CP rule requires for hydrogen mitigation. Is it subsumed by this?

MR. FLEISCHMAN: This rule applies to reactors that were licensed -- whose CPs were issued prior to March 28, 1979. The CP ML rule actually would apply to those with pending construction permits and manufacturing license applications.

In fact, that is why it has been written that way. So the CP ML rule would cover those CPs who came after March 28, 1979. So these are ones that they have already had the construction permit approved.

MR. KERR: How did one reach the conclusion that these ought to be dealt with differently than the CP ML?

MR. FLEISCHMAN: Primarily because construction and design had progressed quite a bit on these plants, and so the feeling was that they didn't want to have an unreasonable ratchet on them. They didn't want to impose ar6

something to make them make significant modifications to their design and yet they wanted to be safe. So there was some sort of compromise between safety and having them redesign their plants completely. The CP ML rule requires 100 percent metal/water reaction to be considered.

MR. KERR: That is what I was thinking. The idea is that this applies to perhaps 99 percent of all plants that are likely to be built in the next 10 years. But if somebody decided to build a new one, at least under current rules, instead of 75, they would have to deal with 100.

MR. FLEISCHMAN: 100. I think he also has to have a three-inch hole for possibly having a vented filtered containment, or things like that.

MR. SIESS: Three foot.

MR. FLEISCHMAN: So for the new plants that haven't come in yet, or that are just pending --

MR. KERR: How much risk reduction is obtained by going from 75 to 100, and from zero to three feet in diameter? Has anybody looked at that?

MR. FLEISCHMAN: I don't know.

MR. SIESS: That is PRA.

MR. FLEISCHMAN: I can't answer you. I don't know.

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I think that is about the end of my presentation.

MR. KERR: Are there any further questions?

Mr. Siess, who is our parliamentarian, tells me that as subcommittee chairman of this subcommittee, I should look at the last letter we wrote on the last hydrogen control rule, we being the ACRS, and see if this differs markedly from that.

MR. FLEISCHMAN: I think I've indicated what the changes were.

MR. KERR: You have not, and you need not have compared it with our letter, and that would be my responsibility. So we may make some comments.

MR. FLEISCHMAN: I appreciate any comments you can make.

MR. SIESS: It is possible that your changes are in the opposite direction to the recommendations last time.

MR. FLEISCHMAN: I don't recall seeing your letter on this.

MR. SIESS: We did not write a letter on the hydrogen control rule at all?

MR. QUITTSCHREIBER: I don't believe so.

MR. KERR: It does appear to me that they are reasonable changes.

MR. SIESS: I assume the Staff does want

comments from the ACRS.

MR. FLEISCHMAN: We would appreciate your comments, yes.

MR. SIESS: It is a question of do you want them.

MR. KERR: You don't want them immediately, as long as we get them in within the comment period.

MR. FLEISCHMAN: The comment period expired. What we intend to do now is go to the CRGR. I think May 11 is the CRGR meeting. We would like to go with them.

MR. SIESS: We haven't signed a memorandum of understanding. At our last meeting there was some attempt at debate of whether we looked at rules before or after CRGR. I think the tendency this time right now is that at the proposed rule stage, before CRGR, but before the final rule stage, we should look at it after CRGR just before the commission sees it and give our advice to the Commission.

MR. FLEISCHMAN: We thought this would be a good time because you were having a meeting and we were ready for you.

MR. KERR: I think this was a good time to make the presentation.

MR. ROSZTOSCY: Mr. Chairman, another approach would be that we would go forward with the rules to the CRGR rules.

MR. KERR: I would say that that would be in accordance with the comments.

MR. SIESS: We wouldn't have any objection, I am sure.

MR. KERR: Any other questions of Mr. Fleischman?

Thank you, sir.

MR. FLEISCHMAN: Thank you very much.

MR. KERR: Any other comments or questions from the Staff about this or anything else we have done today?

Let me thank all of you for your presentations.

Let me say to our consultants, before I close the meeting, that you make some comments in light of what you understand to be regulatory needs. I'm not sure that I could you what they are, but in the context in which you understand them. Whether the activities that are described today, the containment work, the source term work, and the severe fuels work, are appropriate to satisfy regulatory needs, in your view, and whether they are likely to be available on some sort of usable schedule. The first schedule, I guess, being somewhere early in 1984, if I understand it.

Is that sort of the first decision point, or is it later on?

MR. ROSCTOSCZY: I believe the statement in the policy paper which has now been published officially is mid-1984.

MR. KERR: That is 82-1-B, last version?

MR. ROSCTOSCZY: Yes.

MR. KERR: By mid-1984 there should be enough information to make some tentative preliminary decision on a rule.

Then I would want you to concentrate on that. If you have any comments on the perceived objectives of IDCOR or their methods of getting there, we would welcome those. But I look on the IDCOR presentation as more an introductory sort of thing, but I would certainly welcome any comments you have.

Any other comments anybody else wants to make?

This meeting is adjourned.

(Whereupon, at 2:45 p.m., the meeting was adjourned.)

* * * *

CERTIFICATE OF PROCEEDINGS

This is to certify that the attached proceedings before the NRC COMMISSION

In the matter of: ACRS SUBCOMMITTEE ON CLASS 9 ACCIDENTS

Date of Proceeding: Tuesday, April 26, 1983

Place of Proceeding: Washington, D.C.

were held as herein appears, and that this is the original transcript for the file of the Commission.

FRANK G. TAYLOE
Official Reporter - Typed

Frank G. Tayloe
Official Reporter - Signature

STATUS OVERVIEW

ACCIDENT SOURCE TERM PROGRAM OFFICE

(ASTPO)

R. BERNERO, RES/ASTPO

APRIL 26, 1983

ASTPO BASIC AGENDA

- o DOCUMENT CURRENT DATA BASE FOR SEVERE ACCIDENT BEHAVIOR PREDICTION
- o APPLY LATEST BEST ESTIMATE MODELS FOR SEVERE ACCIDENT SOURCE TERMS
- o OBTAIN SUBSTANTIAL AND BROAD PEER REVIEW OF PRINCIPAL WORK
- o SYSTEMATIC EVALUATION OF EMERGENCY PLANNING EXPERIENCE
- o APPLY IMPROVED SOURCE TERM INFORMATION TO REGULATORY PROGRAMS
  - EMERGENCY RESPONSE
  - OTHER

STRATEGY FOR ASTPO ACTION

- o PARALLEL CLOSE-COUPLED EFFORTS
  - REASSESSMENT OF ACCIDENT SOURCE TERMS (NUREG-0772 ---- NUREG-0956)
  - REEVALUATION OF ACCIDENT RISK AND EMERGENCY RESPONSE BASED ON REFINED CURRENT ASSESSMENT AND EMERGENCY RESPONSE EXPERIENCE
  - INCORPORATION OF SOURCE TERM REASSESSMENT AS AVAILABLE AND APPROPRIATE

- o REVIEW AND ACCEPTANCE OF WORK
  - PEER REVIEWS
    - PERIODIC EXPERT PEER REVIEW
    - BROAD SCIENTIFIC PEER REVIEW
  - INTERACTIONS WITH OTHER AGENCIES
    - EPA
    - FEMA
    - STATE RADIATION AUTHORITIES
    - GOVERNORS
    - ETC.
  - ACRS REVIEW

ELEMENTS OF THE REASSESSMENT OF <u>TECHNICAL BASES FOR SOURCE TERMS</u>

<u>ELEMENT 1</u>: SUMMARY OF THE DATA BASE FOR VALIDATION OF CODES TO PREDICT RELEASES

<u>ELEMENT 2</u>: SOURCE TERM ESTIMATES FOR SELECTED PLANTS AND ACCIDENT SEQUENCES

<u>ELEMENT 3</u>: THOROUGH PEER REVIEW OF THE PRECEDING SCIENTIFIC BASIS FOR REASSESSMENT

<u>ELEMENT 4</u>: APPRAISAL OF THE RISK AND REGULATORY SIGNIFICANCE OF REASSESSED SOURCE TERMS

SCHEDULE FOR REASSESSMENT

ELEMENT 1 COMPLETE		7/31/83
ELEMENT 2 COMPLETE		8/31/83
	- BCL SURRY	6/15/83
	- BCL PEACH BOTTOM/GRAND GULF	6/30/83
	- BCL SEQUOYAH	7/15/83
	- BCL SURRY (REVISED)	7/31/83
	- BCL ZION	7/31/83
ELEMENT 3 COMPLETE		12/15/83
ELEMENT 4 COMPLETE		12/15/83
NUREG-0956 PUBLISHED		FEB' 84

PURPOSE OF THE RESEARCH PROGRAM ON SEVERE FUEL DAMAGE

- o TO PROVIDE A DATA BASE AND VERIFIED ANALYTICAL MODELS FOR USE IN ASSESSING THE CONSEQUENCES OF LWR ACCIDENTS INVOLVING SEVERE CORE DAMAGE.

- o APPLICATIONS OF PROGRAM RESULTS ARE TO:

  - REGULATORY DECISIONS FOR ACCIDENT CONDITIONS BEYOND THE DESIGN BASIS.
  - IMPROVED RISK ASSESSMENT METHODOLOGY AND CODES.
  - ASSESSMENT OF POSSIBLE REFINEMENTS IN SYSTEMS AND PROCEDURES.
  - PLANNING FOR SEVERE-ACCIDENT MANAGEMENT, TRAINING, AND EMERGENCY RESPONSE.
  - INFORMATION TO THE PUBLIC AND TO OTHER GOVERNMENT UNITS DURING THE COURSE OF ANY SEVERE ACCIDENT.
  - WHAT ACTUALLY HAPPENED AT TMI-2.

TECHNICAL ISSUES ADDRESSED BY THE
SEVERE FUEL DAMAGE RESEARCH PROGRAM

- FISSION-PRODUCT RELEASE FROM THE CORE, INCLUDING TIMING AND CHEMICAL FORM, AEROSOL FORMATION, AND IN-VESSEL ATTENUATION MECHANISMS.
- HYDROGEN RELEASE FROM THE CORE, INCLUDING TIMING.
- PHYSICAL AND CHEMICAL STATE OF THE CORE DURING SEVERE-ACCIDENT SEQUENCES, INCLUDING THE PROGRESSION OF CORE MELT TO REACTOR-VESSEL FAILURE.
- COOLABILITY LIMITS OF SEVERELY DAMAGED CORES UNDER REFLOOD, INCLUDING REQUIREMENTS ON COOLANT SUPPLY AND TIMING.

INTEGRATED SEVERE FUEL DAMAGE RESEARCH PROGRAM

<u>PURPOSE</u>: TO DEVELOP A DATA BASE AND VERIFIED MECHANISTIC ACCIDENT ANALYSIS MODELS AND CODES THAT INCLUDE THE RISK-SIGNIFICANT PHENOMENA FOR USE IN ASSESSING THE CONSEQUENCES OF ACCIDENTS INVOLVING SEVERE CORE DAMAGE, AND FOR BENCHMARKING RISK-ASSESSMENT CODES. THE MECHANISTIC CODES ARE THE EMBODIMENT OF THE RESULTS OF THE EXPERIMENTAL PROGRAM.

APPROACH:

- o INTEGRAL (MULTI-EFFECT) IN-PILE TESTS - PBF, NRU
  - ESSENTIAL SCOPING ROD-BUNDLE DATA
  - ONLY IN-VESSEL SOURCE TERM DATA
  - PBF PHASE 1, 5 TESTS BY APRIL 1984
  - PBF PHASE 2, NOW ONLY 2 TESTS BY JUNE 1985 - THEN SHUTDOWN PBF
  - NRU - NOW ONLY 2 TESTS TO CHECK FULL-LENGTH EFFECTS
- o SEPARATE EFFECTS PHENOMENOLOGICAL EXPERIMENTS - ACRR AND LABORATORY (KfK)
  - TIME-CONTINUOUS VISUAL DATA FOR FUEL-DAMAGE MODEL DEVELOPMENT
  - VERY COST-EFFECTIVE DATA TO COVER SEVERE ACCIDENT PARAMETER RANGE (BWR)
  - VERIFY LMFBR DEBRIS-COOLABILITY MODELS FOR LWR CONDITIONS (NOW ONLY 3 TESTS)
  - LABORATORY DATA ON ZIRCALOY OXIDATION, FUEL LIQUEFACTION, ETC. (MOSTLY KfK)
- o DEVELOPMENT OF MECHANISTIC MODELS AND CODES
  - SCDAP, MELPROG, TRAC-MIMAS
- o BENCHMARK DATA FROM TMI-2 CORE EXAMINATION

CHANGES TO THE SEVERE FUEL DAMAGE RESEARCH PROGRAM

- o THE DEGRADED CORE COOLABILITY EXPERIMENTS (DCC) IN ACRR WILL BE DISCONTINUED AFTER FY 1983. ONLY THREE EXPERIMENTS WILL BE PERFORMED TO CONFIRM THE APPLICABILITY OF CURRENT LMFBR-BASED COOLABILITY MODELS TO LWR SYSTEMS.

- o THE NRU TEST MATRIX HAS BEEN REDUCED FROM APPROXIMATELY 8 TESTS TO ONLY TWO LENGTH-CONFIRMATION TESTS TO BE COMPLETED BY OCTOBER 1984.

- o THE PBF PHASE II EXPERIMENTAL MATRIX WILL PROBABLY BE REDUCED TO ONLY TWO EXPERIMENTS DUE TO LACK OF FUNDS.

SIGNIFICANT RESULTS OF THE FIRST PBF-SFD EXPERIMENT

- THE FISSION PRODUCT RELEASE RATE DUE TO LIQUEFACTION (I.E., CLAD MELTING AND SOME FUEL DISSOLUTION IN IT) WAS APPROXIMATELY 50 TIMES HIGHER THAN DIFFUSIONAL RELEASE.
- FISSION PRODUCT RELEASE DURING QUENCH WAS APPROXIMATELY 10 TIMES LARGER THAN RELEASE DUE TO LIQUEFACTION AND DIFFUSION.
- IODINE AND TELLURIUM FOLLOWED THE LIQUID PATHWAY. (IMPLIES IODINE IS IN FORM OF CsI AND IS DISSOLVED IN THE WATER.)
- IODINE AND CESIUM RELEASE FROM THE TEST TRAIN WERE ABOUT 20%, AND TELLURIUM RELEASE ABOUT 5%. ALMOST ALL OF THE I_2 AND Cs WAS IN THE BLOWDOWN TANK, AND MOST OF THE Te WAS IN THE FILTER.
- LOW VOLATILITY FISSION PRODUCTS ARE APPARENTLY NOT RELEASED TO ANY SIGNIFICANT EXTENT AT TEMPERATURES BELOW 2400°K (3860°F).
- CURRENT MODELS IN SCDAP CAN ACCURATELY PREDICT THERMAL HISTORY AND LIQUEFACTION OF BUNDLES DURING SEVERE ACCIDENT CONDITIONS. MARCH 1.1 (AS USED BY EG&G) SIGNIFICANTLY OVERESTIMATED LIQUEFACTION AND FUEL TEMPERATURES. HOWEVER, MARCH 2.0 CALCULATIONS AT BCL GAVE SIGNIFICANTLY BETTER AGREEMENT (I.E., ONLY 200-400°K OVERPREDICTION).
- MASS BALANCE OF FISSION PRODUCT SOURCE TERM IS AWAITING EXTENSIVE PIE RESULTS (AUG. 83).

EXPECTED SIGNIFICANT RESULTS OF SECOND PBF EXPERIMENT

- o DETERMINE FISSION PRODUCT RELEASE IN ABSENCE OF QUENCH.
- o DETERMINE HYDROGEN EVOLUTION UNDER FREE-HEATING, STEAM STARVATION CONDITIONS - I.E., CONFIRM EXPECTED DECREASE OF HYDROGEN RELEASE FOR UNATTENUATED CORE BOILOFF CONDITIONS.
- o CONFIRM EXPECTED INCREASE OF FUEL LIQUEFACTION UNDER FAST HEATUP (FREE HEATING) CONDITIONS. DETERMINE SUBSEQUENT EFFECT ON HYDROGEN GENERATION, FUEL RELOCATION, BLOCKAGE FORMATION, DEBRIS CHARACTERIZATION, AND FISSION PRODUCT RELEASE AND TRANSPORT.
- o CONFIRM SCDAP AND MARCH VALIDITY UNDER ABOVE CONDITIONS.
- o PROVIDE DATA FOR MELPROG MODELING.

STATUS OF FOREIGN PARTICIPATION IN SFD PROGRAM

- AGREEMENTS HAVE BEEN SIGNED WITH THE U.K., NETHERLANDS, ITALY, BELGIUM, AND THE FRG.

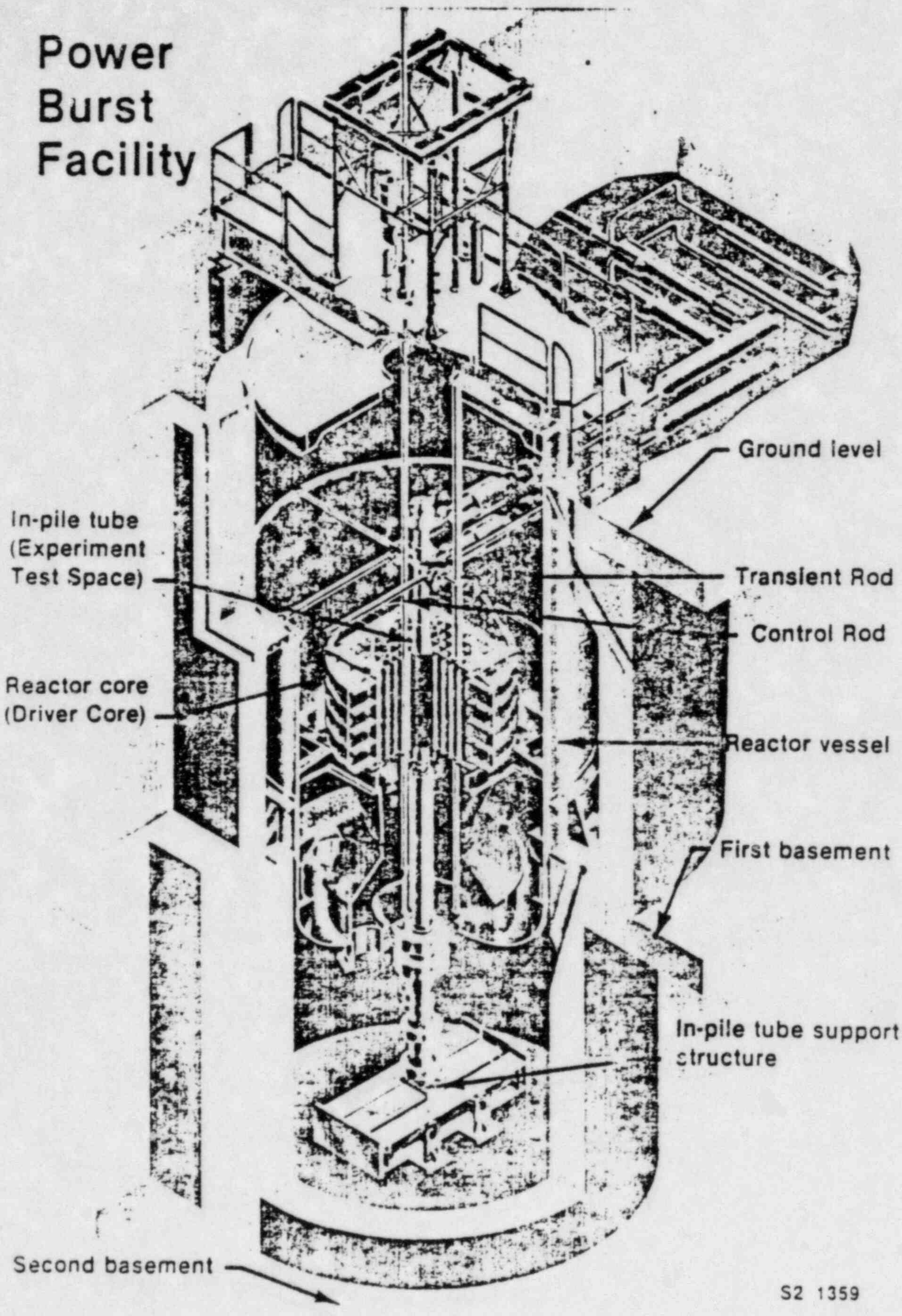
- SIGNING OF AN AGREEMENT WITH JAPAN IS EXPECTED IN MAY (1983).

- NEGOTIATIONS ARE CONTINUING FOR AGREEMENTS WITH CANADA, KOREA, AND TAIWAN.

- TOTAL DOLLAR CONTRIBUTIONS TO THE BEHAVIOR OF DAMAGED FUEL CURRENTLY AMOUNT TO APPROXIMATELY 15%/YR AVERAGED OVER FY 83 AND 84.

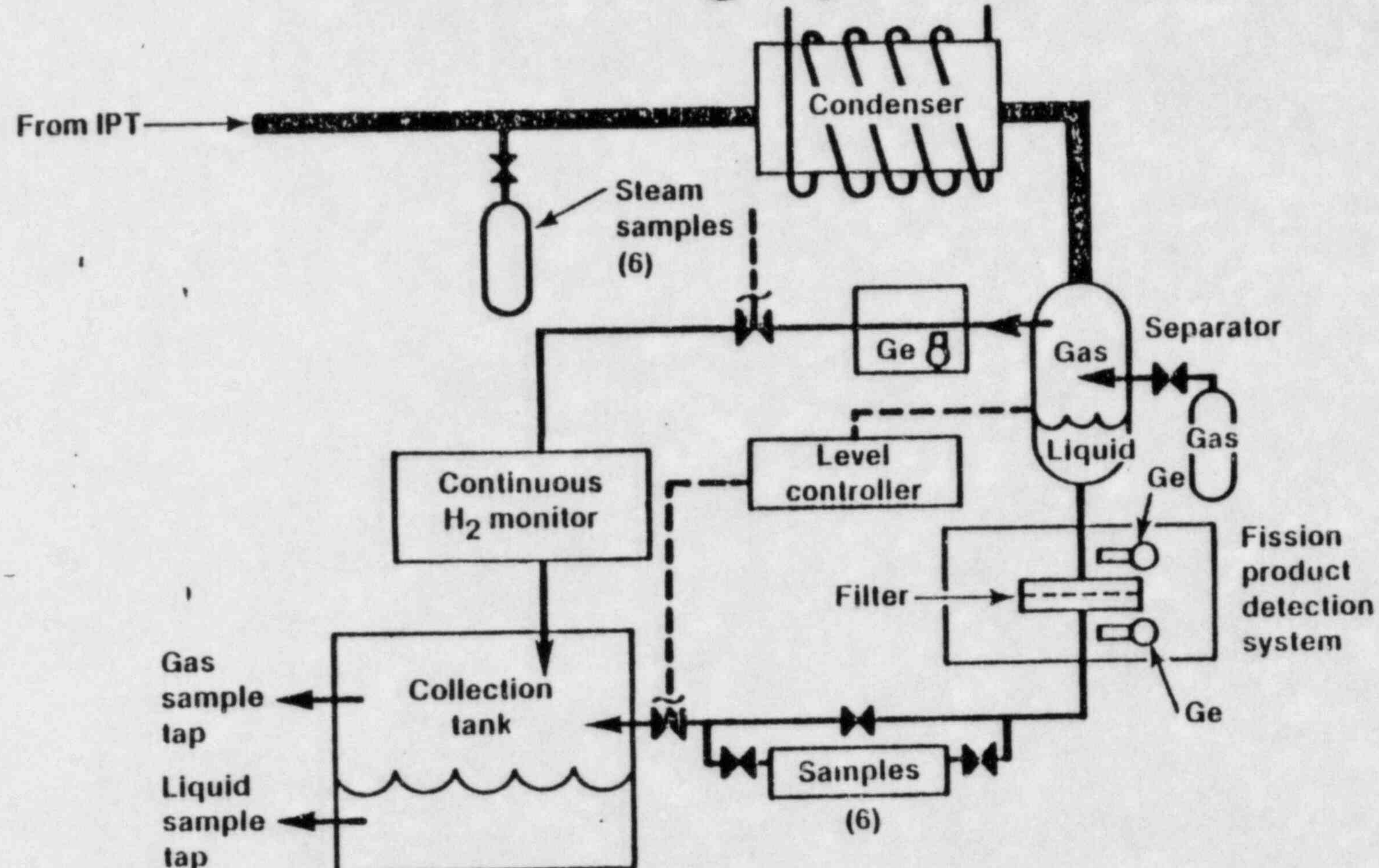
- CONSIDERING IN-KIND RESEARCH CONTRIBUTIONS FROM THE FRG, NETHERLANDS, AND BELGIUM INCREASES THE WORTH TO APPROXIMATELY 25%.

Power Burst Facility



S2 1359

Fission Product and H_2 Monitoring System



INEL-S-35 163

SFD Series 2 Tests

Test Number	Accident Sequence	System Pressure (psi)	Upper Plenum*	Inlet Flow
SFD 2-1	Plant blackout	2300	PWR	Low
SFD 2-2	Interfacing systems LOCA	~100	PWR	0
SFD 2-3	Small break LOCA,ECCs failure	~1000	PWR	Low, variable
SFD 2-4	Transient with scram failure	~1300	BWR	High, variable

* Simplified geometry to provide mechanistic behavior
Peak temperature: ~3100K
Test termination: slow cooling
Test rods: previously irradiated with control rods

TF20230-4

SFD Series I Tests

Test No.	Heating Rate (K/s)	Inlet Flow (g/s)	Test Rods	Cooling	Working Schedule
SFD-ST	< 0.5	13.3	Fresh	Quench	Complete
SFD-1-1	TMI-2*	0.6	Fresh	Slow	04/15/83
SFD-1-2	TMI-2*	0.6	Fresh	Quench	06/30/83
SFD-1-3	TMI-2*	0.6	Irradiated	Slow	01/20/84
SFD-1-4	TMI-2*	0.6	Irradiated Control	Slow	04/23/84
	1K psi				

* Characterized by slow heating rate up to 1600K and rapid heating rate above 1600K, driven by metal-water reaction

TF20228-5

HYDROGEN CONTROL FOR MARK III BWRs

AND ICE CONDENSER PWRs

MORTON R. FLEISHMAN

PRESENTATION FOR THE CLASS-9 ACCIDENT
SUBCOMMITTEE OF THE ACRS

APRIL 26, 1983

RULEMAKING NOTICES

ADVANCE NOTICE - LONG TERM RULE ON DEGRADED CORES (SEVERE ACCIDENTS)	OCTOBER 2, 1980
PROPOSED RULE - HYDROGEN CONTROL AND CERTAIN DEGRADED CORE CONSIDERATIONS	OCTOBER 2, 1980
PROPOSED RULE - PENDING CP/ML APPLICATIONS	MARCH 23, 1981
PROPOSED RULE - PENDING OL APPLICATIONS	MAY 13, 1981
FINAL RULE - INERTING MARK Is & IIs - RECOMBINER CAPABILITY - HIGH POINT VENTS	DECEMBER 2, 1981
PROPOSED RULE - HYDROGEN CONTROL MARK III's & ICE CONDENSERS - EQUIPMENT SURVIVABILITY - ANALYSES	DECEMBER 23, 1981

(CONTINUED)

RULEMAKING NOTICES (CONTINUED)

FINAL RULE - PENDING CP/ML APPLICATIONS	JANUARY 15, 1982
POLICY STATEMENT ON SEVERE ACCIDENTS	APRIL 13, 1983
FINAL RULE - HYDROGEN CONTROL - EQUIPMENT SURVIVABILITY/ QUALIFICATION	1983

FIRST HYDROGEN CONTROL RULE

- INERTING OF MARK I & II BWRs

  EFFECTIVE MAY 4, 1982

  OR

  6 MONTHS AFTER INITIAL CRITICALITY

- EXTERNAL RECOMBINER CAPABILITY (OR INTERNAL RECOMBINERS)

  FIRST SCHEDULED OUTAGE AFTER JULY 5, 1982 OF SUFFICIENT DURATION

- HIGH POINT VENTS

  FIRST SCHEDULED OUTAGE AFTER JULY 1, 1982 OF SUFFICIENT DURATION

STATUS OF PROPOSED RULE

- EXTENDED COMMENT PERIOD EXPIRED APRIL 8, 1982
- DETAILED REVIEW OF COMMENT COMPLETED
  - 28 PERSONS SUBMITTED COMMENTS
- COMMISSION PAPER ON FINAL RULE IN PREPARATION

(CONTINUED)

STATUS OF PROPOSED RULE (CONTINUED)

- CHANGES UNDER CONSIDERATION
  - DEFER REQUIREMENTS ON DRY CONTAINMENTS UNTIL SEVERE ACCIDENT DECISION
  - REVISE IMPLEMENTATION SCHEDULE
  - PERMIT OTHER ACCEPTABLE METHODS FOR SHOWING CONTAINMENT INTEGRITY
  - ELIMINATE REQUIREMENT FOR COLD SHUTDOWN
  - ALLOW SHOWING THAT LOCAL DETONATIONS CANNOT OCCUR
  - ELIMINATE NEED FOR COMPARATIVE ANALYSIS OF ALTERNATIVE CONTROL SYSTEMS
  - PERMIT USE OF DIFFERENT METHODS OF ACCIDENT ANALYSIS

FINAL RULE

- H_2 CONTROL FOR MARK III, ICE CONDENSERS

  EFFECTIVE [~~1~~] 2 YEARS AFTER EFFECTIVE DATE

  OR

  LICENSE ABOVE 5 PERCENT POWER

- EQUIPMENT SURVIVABILITY DURING H_2 BURN

  MARK III AND ICE CONDENSERS

  \- EFFECTIVE [~~1~~] 2 YEARS AFTER EFFECTIVE DATE

  [~~OTHER LWR'S~~

  ---~~EFFECTIVE 2 YEARS AFTER EFFECTIVE DATE~~]

- ANALYSES

  \- EFFECTIVE 1 YEAR AFTER EFFECTIVE DATE

  H_2 CONTROL FOR MARK III, ICE CONDENSERS

  CONTAINMENT STRUCTURAL INTEGRITY AND FUNCTIONING OF SYSTEMS AND COMPONENTS DURING A H_2 BURN FOR MARK III, ICE CONDENSERS.

FINAL RULE REQUIREMENTS

- HYDROGEN CONTROL SYSTEMS FOR MARK III AND ICE CONDENSER PLANTS

  - 75% FUEL CLADDING - WATER REACTION
  - NO LOSS OF CONTAINMENT INTEGRITY BY ACCEPTED METHOD
  - ACTUAL MATERIAL PROPERTIES WITH MARGINS, OR
  - ASME SERVICE LEVEL C OR FACTORED LOAD CATEGORY LIMITS

- FUNCTIONING OF SYSTEMS AND COMPONENTS DURING HYDROGEN BURN FOR MARK IIIs AND ICE CONDENSERS ~~NON-INERTED-LWR'S~~

  - NEEDED TO ESTABLISH AND MAINTAIN SAFE [~~COLD~~] SHUTDOWN AND CONTAINMENT INTEGRITY
  - BURNING OF HYDROGEN
  - 75% FUEL CLADDING - WATER REACTION
  - LOCAL DETONATIONS INCLUDED UNLESS UNLIKELY TO OCCUR

- ANALYSES FOR ABOVE REACTOR CATEGORIES

  - [~~JUSTIFY~~] SUPPORT SELECTION OF HYDROGEN CONTROL SYSTEM
  - ASSURE CONTAINMENT STRUCTURAL INTEGRITY
  - ASSURE FUNCTIONING OF CERTAIN SYSTEMS AND COMPONENTS

Marii 4/25/83 mf

7/10/83

mf

<u>For</u>: The Commissioners

<u>From</u>: William J. Dircks, Executive Director for Operations

<u>Subject</u>: AMENDMENTS TO 10 CFR PART 50 RELATED TO HYDROGEN CONTROL

<u>Purpose</u>: To obtain Commission approval for publication of final amendments in the Federal Register.

<u>Category</u>: This paper covers a major policy question.

<u>Issue</u>: Whether applicants and licensees with Mark III BWRs and PWR ice condenser facilities should be required to:

a. Provide hydrogen control systems that can handle large amounts of hydrogen,

b. Demonstrate the survivability/qualification of containment and safety systems during and following a hydrogen burn, and

c. Perform and submit analyses concerning hydrogen control and survivability/qualification of containment and safety systems.

<u>Discussion</u>: During the Policy Session on September 16, 1981, the Commission was briefed by the staff on Interim Amendments to 10 CFR Part 50 Related to Hydrogen Control (SECY 81-245A). The discussion covered both a final and a proposed rule and resulted in several Commission comments that required resolution. The Commission approved publication of the final rule during Affirmation Session 81-41 on November 5, 1981 (Enclosure "A"). The final rule was published in the Federal Register on December 2, 1981 (46 FR 58484) and required inerted atmospheres for BWR Mark I and II containments, hydrogen recombiner capability for LWRs that rely on purge/repressurization systems as the primary means of hydrogen control, and high point vents

Contact:
M. Fleishman, RES
443-5997

for all LWRs. The Commission approved publication of the proposed rule during Affirmation Session 81-43 on November 24, 1981 (Enclosure "B").

The proposed rule (Enclosure "C") was published in the Federal Register on December 23, 1981 (46 FR 62281), and allowed 60 days for a public comment period which expired on February 22, 1982. A notice of extension of comment period (Enclosure "D"), including editorial corrections, was published on February 25, 1982 (47 FR 8203) and extended the comment period for an extra 45 days to April 8, 1982. The proposed rule would have required that:

a. Each boiling water reactor with a Mark III type containment and each pressurized water reactor with an ice condenser type containment be provided with a hydrogen control system capable of handling an amount of hydrogen, equivalent to that which would be generated if there were at least a 75 percent fuel cladding-water reaction, without loss of containment integrity;

b. Each boiling water reactor and each pressurized water reactor that does not rely on an inerted atmosphere for hydrogen control be provided with safety systems, needed to establish and maintain safe cold shutdown and maintain containment integrity, that can function after the burning of substantial amounts of hydrogen; and

c. Analyses be performed for the reactor categories mentioned above to justify the hydrogen control systems selected and to assure containment structural integrity and survivability of needed safety systems during a hydrogen burn.

In response to the notice of proposed rulemaking, comments were submitted by 28 persons having the following affiliation:

Nuclear Steam System Suppliers	3
Utilities	18
Architect/Engineer Firms	2
Industrial Associations	3
Individuals	2

A detailed summary of the comments is provided in Enclosure "E", including a list of commenters, and a paraphrase of each of 202 comments. The comments received covered all aspects of the proposed rule and there was a considerable amount of duplication among commenters. The following represents a distillation and paraphrasing of the more significant comments:

1. The implementation of the Hydrogen Control Rule should be deferred until the severe accident rulemaking when applicable research and probabilistic risk analyses (PRAs) will be completed.

   Resolution: The staff agrees with these comments relative to PWRs with large dry containments. Because of the greater inherent capability of the dry containment designs to accommodate large quantities of hydrogen (higher design pressure and larger volume), the staff believes that rulemaking with regard to hydrogen control can be safely deferred pending completion of NRC- and industry-sponsored research. With regard to systems and components that must be able to function during and following hydrogen burning, the results of the TMI-2 containment survey indicates that such systems and components did function properly following the burn event.

   With regard to BWRs with Mark III containments and PWRs with ice condenser containments, the staff believes that the rulemaking should be carried forward. This will formalize Commission regulatory decisions currently being applied on a case-by-case basis.

2. The 75 percent metal-water reaction required to be assumed for design and analysis is unreasonably high based on evaluation of the TMI-2 accident and analyses of recoverable degraded core accidents.

   Resolution: The staff agrees that the 75 percent metal-water reaction is significantly greater than that which occurred during the TMI-2 accident. However, the primary intent of the rule is to require containment designs that can accommodate accident sequences in which hydrogen combustion poses the principal threat to containment integrity. Consequently, the staff believes it is prudent to specify a value sufficiently greater than that which was analyzed to have occurred at TMI-2 so that there will be an appropriate margin of safety. In this regard, it should be noted that the 75 percent value refers only to the cladding surrounding the active fuel region. Not all of the zirconium which can interact is in this fuel cladding. For example, BWR channel box temperatures may be close to the cladding temperature, just as the grid spacers in PWRs and BWRs will be. All these contain zirconium, and the intent of the 75 percent value is to account for reactions in these items as well. The staff feels confident that the 75 percent value is representative of a limiting case degraded core accident. Finally, the staff sees no significant benefit in reducing the metal-water reaction to a level such as 50 percent for those plants required to install a hydrogen control system since the basic design of the system would not change.

3. The requirement for a hydrogen control system should be revised to permit licensees the option of analytically demonstrating that additional hydrogen control systems are not necessary because of intrinsic design features that reduce the likelihood of hydrogen generation.

   Resolution: While the staff agrees that design features to reduce hydrogen generation are necessary and desirable, it still believes that, in order to cope with unexpected events, there should be a solution to the hydrogen issue that involves design features that ensure containment integrity, even if a large amount of hydrogen is generated.

4. Since the primary function of the containment is to prevent excessive radiation dose to the public, the rule should be modified to preclude the loss of containment function rather than to preclude the loss of containment integrity.

   Resolution: The staff appreciates the fact that some nuclear plants are designed with a multi-building, multi-barrier concept that is intended to prevent the leakage of radiation by diverse methods such as filtering or scrubbing mechanisms, plate-out mechanisms and containment sprays. However, the Commission's basic and long-standing safety philosophy has been that the containment should be designed to remain intact following an accident in order to provide additional assurance that excessive radiation will not be released. The staff supports this policy that the prevention of excessive radiation dose to the public can best be assured by maintaining a leak tight containment; and that this, in turn, can be provided by assuring that there is structural integrity with margin.

5. The criterion for containment structural integrity is unnecessarily restrictive. It should not be limited to the provisions of the ASME Boiler and Pressure Vessel Code, but should permit other methods such as realistic analyses using actual material properties.

   Resolution: The staff agrees with this comment and has modified the rule in this regard. The rule has been changed to indicate that "containment structural integrity must be demonstrated by use of a method ~~previously~~ accepted by the NRC staff." The rule includes ~~two~~ alternative methods as examples but does not preclude other methods that may be shown to be acceptable to the Commission. /

   that has been

6. The rule should address only non-inerted, small-volume, low-pressure containments and should not impose requirements on the remaining containments since it would provide, at best, insignificant improvements in safety.

   Resolution: The staff agrees for the reasons indicated above and has, accordingly, revised the rule to apply only to Mark III BWRs and ice condenser PWRs.

The staff does not agree that the post-TMI improvements have been ignored. However, with respect to

7. The rule ignores those post-TMI suggested improvements which have been implemented and which reduce the likelihood of a degraded core accident.

   Resolution: ~~The staff agrees with this comment. In the case of~~ PWRs with large dry containments, the staff feels that the post-TMI improvements, along with the inherent strength of the containments, have indeed provided sufficient safety to permit the delay of any additional rulemaking until completion of ongoing research programs.

8. In view of the small probability of occurrence of local detonations as a result of various design features, the rule should permit licensees the option of demonstrating that local detonations cannot occur in lieu of evaluating the effects of local detonations.

   Resolution: The staff agrees with this comment and has modified the rule appropriately.

9. The requirement that systems and components be provided for safe cold shutdown is unnecessary and is inconsistent with the licensing basis for most operating plants which requires only safe shutdown. It should not be an issue with regard to hydrogen control but should be considered in another forum.

   Resolution: The staff agrees with this comment and has modified the rule appropriately. Because of the fact that a degraded core accident is less likely than a design basis accident, the staff believes that the requirement for cold shutdown may be overly conservative. The licensing basis for most plants is, in fact, just safe shutdown. The issue of safe shutdown versus safe cold shutdown is expected to be addressed within the context of the resolution of Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," which is scheduled for completion by October 1984.

10. The implementation schedules should be made more realistic so that design changes logically follow after the required analyses are completed.

Resolution: The staff agrees. The greatest relief, of course, has come by deferring implementation of the rule for PWRs with large dry containments. However, the rule has also been revised to specify that the required analyses be submitted to the Commission within 1 year and the corresponding design changes be completed within 2 years.

11. In the Supplementary Information accompanying the rule, it was stated that the selection of the hydrogen control system should be supported by comparative analyses of alternative systems to show their relative advantages and disadvantages. This guidance is inconsistent with Commission practice and is unnecessary. The only requirement should be a demonstration that the selected system is suitable for its intended application.

    Resolution: The staff agrees that this is inconsistent with Commission practice in the case of NTOLs and ORs and has modified the guidance accordingly. The rule has also been modified to delete the implication that comparative analyses are required and to indicate that the analysis is intended to support the design of the hydrogen control system selected.

12. The two-step approach to equipment survivability, described in the Supplementary Information section of the notice of proposed rulemaking, is unwarranted and will unnecessarily escalate the costs to industry.

    Resolution: The staff agrees with this comment, particularly in view of the smaller likelihood of a degraded core accident as compared to a design basis accident; this has been reduced further by post-TMI improvements. The Commission requested comments on the two-step approach when the proposed rule was issued. The consensus of the comments received was overwhelmingly against the two-step approach. Many commenters felt that a straightforward survivability approach would be appropriate provided reasonable criteria are specified. The staff now believes, in view of the recent issuance of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety," 74 there is no significant difference between demonstrating survivability and demonstrating qualification. Paragraph (f) of § 50.49 describes several methods, one of which must be used, for qualifying electrical equipment important to safety. For example, for those licensees which have already demonstrated survivability, as described in the Supplementary Information of the proposed rule, the same type of qualification methods given in paragraphs (f)(2) and (f)(4) of § 50.49 could be used to show that the systems and components have been qualified. In this regard, the margins considered adequate for a degraded core accident are less than those considered adequate for a design basis accident due to the lower probability of occurrence of a degraded core accident. The staff now views "qualification" as the generation and maintenance of evidence using tests and analyses to assure that systems and components will operate on demand to meet system performance requirements. In the case of a hydrogen burn environment, this means that there must be evidence that systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity are capable of performing their functions during and after exposure to the environmental conditions created by the postulated accident including the burning of hydrogen. Qualification may be demonstrated in a manner acceptable to the staff using a combined approach of analysis and testing. Thus, an acceptable thermal analysis would have to be performed for the containment in order to determine the thermal response of the ~~systems and~~ components during a hydrogen burn. This thermal response would then be compared to the thermal response the ~~systems and~~ components had during their qualification testing. The licensee would then demonstrate that the qualification thermal response envelops the thermal response during a hydrogen burn. Selected tests would also be performed at predicted hydrogen burn conditions (or, other tests previously performed may be referenced if demonstrated to be applicable) to convince the staff that the systems and components are qualified to perform their functions during and following a hydrogen burn.

Along with the proposed rule, the Commission included a description of three different approaches concerning the supplementary guidance to be provided for performing the required analyses for the design of the hydrogen control system. These were (a) analyses of different accident scenarios, (b) analyses of a single accident scenario with variation of key parameters, and (c) analyses using an "envelope of time histories of hydrogen and steam release rates" to be supplied by the Commission. The Commission requested comments concerning which of the approaches was preferred as well as suggestions regarding improvements or other alternatives.

There was no preponderance of comments leaning toward a particular approach; however, the first two approaches appeared to have greater support. Furthermore, many commenters felt that there should be flexibility in the approach to be used and in the selection of the accident scenarios. It was also suggested that the accident scenarios should be considered in order of importance using PRA techniques.

Based on the comments received, the staff is recommending that the Commission not choose between the first two approaches and that licensees need not use the third approach. It should be left to each licensee to suggest to the Commission which of the first two approaches it wishes to use and to arrive at a mutually agreeable method with the Commission for performing the analyses.

The above and all other suggestions from the commenters were reviewed and considered by the staff in preparing the final rule. The final rule, included in the Federal Register notice (Enclosure "F") incorporates changes that reflect the above discussed resolutions and the other comments that were received. The regulation has been printed in comparative text for ease in identifying the changes. A Regulatory Analysis of the final rule is provided by Enclosure "G".

The major changes in the rule from those originally proposed are as follows:

1. The rule has been restricted to Mark III BWRs and ice condenser PWRs with rulemaking for LWRs with large dry containments deferred to the time of the severe accident rulemaking decision.

2. The implementation schedule has been revised to require only the analyses in 1 year; the corresponding design changes would not be required until 2 years.

3. The method for demonstration of containment structural integrity has been revised to broaden the options available. It is indicated that a method ~~acceptable~~ that has been accepted by ~~to~~ the NRC staff is required rather than limiting consideration only to the ASME Boiler and Pressure Vessel Code. The code is included as an example of one of the acceptable methods.

4. The requirement for systems and components that must be able to function following a hydrogen burn has been revised to include "safe shutdown" rather than "safe cold shutdown."

5. The requirement to include the effect of local detonations has been modified so that they would not have to be included if it is shown that local detonations are unlikely to occur.

6. The rule has been modified to eliminate the need for comparative analysis of alternative hydrogen control systems. The rule now indicates that the analyses only have to support the design of the selected hydrogen control system.

Recommendations: That the Commission:

1. Approve the publication of final amendments, as set forth in Enclosure "F", which would require for Mark III BWRs and ice condenser PWRs, hydrogen control systems, assurance of containment structural integrity and systems and components that can perform their functions during and following a hydrogen burn, and supporting analyses.

2. Note:

   a. That these amendments are applicable to Mark III BWRs and ice condenser PWRs whose CPs were issued prior to March 28, 1979. Other related amendments pertaining to applicants with pending CP and manufacturing license applications were published on January 15, 1982 and are also described in NUREG-0718, Rev. 1, dated July 14, 1981. Requirements for future generations of LWRs are under development.

   b. That the notice of final rulemaking in Enclosure "F" will be published in the Federal Register to be effective 30 days after publication.

   c. That pursuant to § 51.5(d) of Part 51 of the Commission's regulations, neither an environmental impact statement nor a negative declaration need be prepared in connection with the amendment since the amendment is nonsubstantive and insignificant from the standpoint of environmental impact.

   d. The reporting requirements in connection with the analyses required by the rule (Enclosure "F") impose information collection requirements that are subject to the Paperwork Reduction Act. The requirements were ~~submitted to the~~ approved by OMB. ~~for review and approval.~~

   e. That pursuant to the Regulatory Flexibility Act of 1980 the rule contains a statement that the Commission certifies that the rule will not, if promulgated, have a significant economic impact upon a substantial number of small entities and a copy of this certification will be forwarded to the Chief Counsel for Advocacy, SBA by the Division of Rules and Records, ADM.

f. That the Subcommittee on Nuclear Regulation of the Senate Committee on Environment and Public Works, the Subcommittee on Energy and the Environment of the House Committee on Interior and Insular Affairs, the Subcommittee on Energy Conservation and Power of the House Committee on Energy and Commerce, and the Subcommittee on Environment, Energy and Natural Resources of the House Committee on Government Operations will be informed.

g. That a Regulatory Analysis is attached as Enclosure "G".

h. That a public announcement will be issued (Enclosure "H").

i. That copies of the Notice of Final Rulemaking will be distributed by TIDC, ADM to each affected licensee and other interested parties.

j. That the staff recommends the paper be placed in the PDR.

Scheduling: Recommend affirmation at an open meeting. No specific circumstance is known to the staff which would require Commission action by any particular date in the near term.

William J. Dircks
Executive Director for Operations

Enclosures:
"A" - Memorandum Chilk to Dircks, dtd 11/6/81
"B" - Memorandum Chilk to Dircks, dtd 11/27/81
"C" - Notice of Proposed Rulemaking
"D" - Notice of Extension of Comment Period
"E" - Summary of Public Comments on Proposed Amendments
"F" - Notice of Final Rulemaking
"G" - Regulatory Analysis
"H" - Draft Public Announcement

REFER TO: M811105B

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

OFFICE OF THE SECRETARY

November 6, 1981

MEMORANDUM FOR: William J. Dircks, Executive Director for Operations
Leonard Bickwit, Jr., General Counsel
Forrest Remick, Director, Policy Evaluation

FROM: Samuel J. Chilk, Secretary

SUBJECT: STAFF REQUIREMENTS - AFFIRMATION SESSION 81-41, 3:00 P.M., THURSDAY, NOVEMBER 5, 1981, COMMISSIONERS' CONFERENCE ROOM, D.C. OFFICE (OPEN TO PUBLIC ATTENDANCE)

I. Draft Order for Oral Presentation in the Waste Confidence Proceeding

The Commission unanimously approved an order specifying procedures for oral presentations to the Commission in the waste confidence proceeding. A majority of the Commission (Commissioners Gilinsky and Bradford disapproving) voted to delete Item 3 on page 13 of the order, which invited comment on the generic subject of accident waste disposal and specifically on the nuclear waste resulting from the TMI-2 accident. (OPE)

(Subsequently, the Order was signed by the Secretary.)

II. NFS Request for a Stay of a Hearing on License Amendment to West Valley License

The Commission, by a vote of 3-1 (Commissioner Ahearne dissenting and Commissioner Roberts abstaining), approved an order which denies NFS's motion for a stay of the license amendment and instructs the ASLBP to initiate a proceeding on the request for a hearing. Commissioner Ahearne's separate views will be included in the order. (OGC)

(Subsequently, the Order was signed by the Secretary.)

III. SECY-81-245A - Interim Amendments to 10 CFR Part 50 Related to Hydrogen Control

The Commission unanimously approved for publication in the Federal Register a final rule to require inerted atmospheres for BWR Mark I and II containments and hydrogen recombiner capability for LWRs that rely on purge/repressurization systems as the primary means of hydrogen control. (RES) (SECY Suspense: 11/20/81)

Enclosure "A"

The Commission requested that:

1. the appropriate Congressional committees be informed; (RES) (SECY Suspense: 11/20/81)

2. a public announcement be issued; (OPA/RES) (SECY Suspense: 11/20/81)

3. notices of the final rule be distributed to affected licensees and other interested parties. (ADMIN/RES) (SECY Suspense: 11/20/81)

A proposed rule on hydrogen control in Mark III and ice condenser containments will be acted upon at a later date.

cc: Chairman Palladino
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
OPA
Public Document Room

Enclosure "A"

to the NRC of significant events that occur at operating nuclear power plants. The Commission requested that:

1. The appropriate Congressional committees be informed;

2. a copy of the FRN be sent to all applicants, licensees and State Governments; and

3. the information collection requirements of this proposed rule be submitted to the OMB for review under the Paperwork Reduction Act. (RES) (SECY Suspense: 12/14/81)

III. SECY-81-619 - Request for Hearing on Big Rock Point

The Commission unanimously approved issuance of an Order denying the request for a hearing. By a vote of 3 to 2 (Commissioners Gilinsky and Bradford disapproving), a majority of the Commissioners denied staff review of a separate safety concern regarding the location of the spent fuel pool and reactor vessel within the same containment. (OGC)

(Subsequently, the Order was signed by the Secretary.)

IV. SECY-81-620 - Request for Hearing on Turkey Point

The Commission unanimously approved an Order denying a request for a hearing for which opportunity had been offered in a confirmatory Order of the Director, Division of Licensing, NRR, imposing certain requirements related to the TMI Action Plan on Florida Power & Light Company's Turkey Point plant. (OGC)

(Subsequently, the Order was signed by the Secretary.)

V. SECY-81-632 - Amendments to Part 2 (Express Mail; Oral Responses to Motions to Compel)

The Commission unanimously approved for publication in the Federal Register final amendments to Part 2 that permit licensing boards to require that answers to motions to compel responses to discovery be provided orally. (OGC)

(Subsequently, the Order was signed by the Secretary.)

cc: Chairman Palladino
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
Commission Staff Offices
Public Document Room

7 CFR Part 1135

[Docket No. AO-380-A1]

Milk in the Southwestern Idaho-Eastern Oregon Marketing Area; Decision on Proposed Amendments to Marketing Agreement and Order

Correction

In FR Doc. 81-36068, appearing at page 61480 in the issue of Thursday, December 17, 1981, the citation in parentheses in lines 12 and 13 of the second paragraph of column two on page 61480 should have read, "(46 FR 32873)".

BILLING CODE 1505-01-M

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Interim Requirements Related to Hydrogen Control

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission is considering amending its regulations to improve hydrogen control capability during and following an accident in light-water reactor facilities.

The amendments would require improved hydrogen control systems for boiling water reactors with Mark III type containments and for pressurized water reactors with ice condenser type containments. All light-water nuclear power reactors not relying upon an inerted atmosphere for hydrogen control would be required to show that certain important safety systems must be able to function during and following hydrogen burning.

DATES: Comment period expires February 22, 1982. Comments received after that date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before that date.

FOR FURTHER INFORMATION CONTACT: Morton R. Fleishman, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone 301-443-5981.

ADDRESS: Written comments or suggestions for considertion in connection with the proposed amendments should be submitted to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. Copies of comments received may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

SUPPLEMENTARY INFORMATION: The accident at Three Mile Island, Unit 2 (TMI-2) resulted in a severely damaged or degraded reactor core, a concomitant release of radioactive material to the primary coolant system, and a fuel cladding-water reaction which resulted in the generation of a large amount of hydrogen. The Nuclear Regulatory Commission has taken numerous actions to correct the design and operational limitations revealed by the accident. Included in these actions are several rulemaking proceedings intended to improve the hydrogen control capability of light-water nuclear power reactors. On October 2, 1980, the Nuclear Regulatory Commission published in the Federal Register (45 FR 65466) a notice of proposed rulemaking on "Interim Requirements Related to Hydrogen Control And Certain Degraded Core Considerations" (Interim Rule). The notice concerned proposed amendments to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to improve hydrogen management in light-water reactor facilities and to provide specific design and other requirements to mitigate the consequences of accidents resulting in a degraded reactor core.

On March 23, 1981, the Commission published in the Federal Register (46 FR 18045) a notice of proposed rulemaking on "Licensing Requirements for Pending Construction Permit and Manufacturing License Applications." The notice proposed a set of licensing requirements applicable to construction permit applications that stemmed from lessons learned from the TMI-2 accident. On May 13, 1981, the Commission published in the Federal Register (46 FR 26491) a notice of proposed rulemaking on "Licensing Requirements for Pending Operating License Applications" (OL Rule).

As a result of the various activities and considerations relative to the October 2, 1980 notice, the Commission decided to split the Interim Rule into two parts. One part was to be included in the OL Rule. The other part, limited only to hydrogen control, was to be issued separately. The details of this split are described in the companion Federal Register notice published on December 2, 1981 (46 FR 58484) concerning hydrogen control related to inerting, hydrogen recombiner capability and high point vents.

The Commission has also been considering the ability of all light-water reactors, particularly pressurized light-water reactor facilities with ice condenser type containments and boiling light-water reactor facilities with Mark III type containment, to withstand an accident with the concomitant generation of large amounts of hydrogen, such as the type which occurred at Three Mile Island, Unit 2 (TMI-2). As a result, three new amendments to the regulations are being proposed for public comment.

Hydrogen Control for Mark III BWRs and Ice Condenser PWRs [§ 50.44(c)(3)(iv)]

It is proposed that boiling water reactor (BWR) facilities with Mark III type containments and pressurized water reactor (PWR) facilities with ice condenser type containments, for which construction permits were issued prior to March 28, 1979, be required to install hydrogen control systems capable of accommodating an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding (surrounding the active fuel region) with water, without loss of containment integrity. This new requirement is being contemplated as a result of safety issues raised during licensing reviews of new ice condenser and Mark III plants. In these reviews, it has become clear that additional protection is required to provide assurance that large amounts of hydrogen can be safety accommodated by these plants. The particular type of hydrogen control system to be selected is left to the discretion of the applicant or licensee; however, it must be found acceptable by the NRC based upon suitable programs of experiment and analysis. The selection should be supported by comparative analyses of alternative systems to show their relative advantages and disadvantages. These comparisons are to be submitted as part of the analyses required under § 50.44(c)(3)(vi). At present, a distributed igniter system has been found acceptable for the Sequoyah plant with an ice condenser containment, but only as an interim solution while the hydrogen control matter is studied further. A post-accident inerting system has also been discussed for the ice condenser and Mark III containments. Whatever systems are finally proposed and approved for the long term, large amounts of hydrogen must be safely accommodated, and operation of the system, either intentionally or inadvertently, must not further aggravate the course of an accident or endanger the plant during normal operations. The amount of hydrogen to be assumed in the design of the

Enclosure "C"

hydrogen control system is that amount generated by assuming that 75% of the fuel cladding surrounding the active fuel region reacts with water. The 75% is judged to be representative of the maximum amount of hydrogen likely to be generated in an accident in which the threat to the containment is limited to the threat posed by the combustion of hydrogen. Events with metal-water reactions in excess of 75% are judged to be associated with core-melt accidents which could pose a threat to containment greater than the combustion of hydrogen. This 75% value also appears to be reasonable because it is sufficiently greater than the fuel cladding-water reaction analyzed to have occurred at TMI-2 to provide a conservative estimate for the cladding reaction that may occur during a TMI type degraded core accident. It is expected that the 75% value will permit plants that are either completed or are well along in the construction stage to have a hydrogen control system added without the need for major modifications to their containment structures. Research now in place will, over the next several years, yield data on the likelihood of termination of sequences with large amounts of cladding interaction.

The Commission would particularly welcome comments on whether the percent of fuel cladding that reacts with water should be less than, equal to, or greater than the 75 percent value being proposed for use in the rules covered by this notice. Supporting analyses, as available, would also be welcome.

Owners of Mark III BWR's now under construction have been surveyed by the NRC staff to determine the effect on their plant designs of the requirement that they do not exceed ASME Service Level A Limits or the Service Load Category during inadvertent full inerting of a post-accident inerting system. This survey was conducted because a post-accident inerting system (rather than a distributed ignition system) was thought to be the preferred approach for the Mark III containments. Based on their responses, the Commission has concluded that there would be no significant impact in specifying these requirements for inadvertent full inerting. Modest deviations from these ASME criteria will be permitted if good cause is shown. A comparable survey was not conducted for ice condenser plants because the distributed ignition system apparently is the approach preferred by the owners of these plants.

There are ongoing programs of reasearch in a number of areas of hydrogen generation, release, burning, and control. These include the analysis of accident sequences, the chronology of hydrogen and steam injection (from the primary system into containment), the analysis of operations to recover coolability, and an assessment of equipment survivability. These studies are expected to reveal the advantages and disadvantages of various hydrogen control systems, including those that involve deliberate burning of the hydrogen within containment. Based on the state of technology as of August 1981, the Commission believes that control methods that do not involve burning provide protection for a wider spectrum of accidents than do those that involve burning.

As a result of the review of the deliberate ignition systems installed at Sequoyah and McGuire, the staff has identified issues which need to be investigated further. A spectrum of degraded core accident scenarios, including those which may lead to inadvertent suppression of combustion in the lower compartment due to a steam rich atmosphere, and several hydrogen combustion phenomena are continuing to be reviewed. In addition, there is incomplete verification of analytical models and equipment survivability. These issues are being addressed in ongoing research by NRC and the nuclear industry. The Commission concludes, based on available information, that the issues are sufficiently resolved to warrant interim approval of deliberate ignition systems for ice condenser plants. However, the Commission has required in individual licensing proceedings and in the section of this rule on analyses (§ 50.44(c)(3)(vi)) that studies of alternative hydrogen management systems be performed prior to the long-term approval of any particular method.

Standards for Safety Systems and Components That Must Function During or After Hydrogen Burn [Sec. 50.44(c)(3)(v)]

The Commission is considering a two-step approach to address qualification of essential equipment during and after a hydrogen burn. As a first step, essential equipment must be demonstrated to "survive" the hydrogen burn and continue to be able to perform its safety function. In this context, the equipment would not have to meet the more rigorous standards of the NRC's equipment qualification program but a different standard as defined below. As a second step, the Commission would require "qualification" of essential equipment.

The Commission feels a two-step approach is justified in light of our lack of knowledge of the probabilities of hydrogen-producing accident scenarios, the environmental conditions during a hydrogen burn, and the effect this environment has on different equipment. The Commission will develop "survivability" criteria which are intended as an interim step to assure the quality of essential equipment until enough information is accumulated from ongoing research to suitably define what equipment performance standards are appropriate. After sufficient information is developed, the Commission may propose long-term standards that are more stringent than the short-term or "survivability" standard being proposed.

The differences in concept between equipment demonstrated to meet the "survivability" standard and equipment that meets the "qualification" standard are described below. The Commission specifically seeks comment on the use of the two step approach for defining equipment standards, the "survivability" and "qualification" standards themselves, and proposals for implementation schedules developed on a well informed basis. Equipment required to be qualified (Eq) and equipment for which survivability must be demonstrated (Es) can be compared as follows:

(a) *Environmental Conditions*—The environmental conditions under which Eq must operate would be calculated using a model that has been demonstrated to be conservative by comparison with numerous experiments and by a long history of usage. For Es, the calculational model contains some conservatisms, but the level of assurance is generally not comparable to that for the Eq model due to a lack of available experimental data for verification.

b. *Testing Conditions*—For Eq, the test conditions would be more severe than the environmental conditions due to extra margins added to account for uncertainties in the test environment, inaccuracies of the measuring devices, variability of the test specimens, etc. For Es, the test conditions need not provide margin beyond the conservatively calculated environmental conditions.

c. *Operability*—Eq and Es would both be required to perform their functions during and after being exposed to their respective test conditions.

d. *Performance*—During and following a test, Eq would be required to perform to specifications determined by accident analyses performed prior to the test; however, for Es, a relaxation of these specifications would be permitted, as defined on a case-by-case (e.g., more instrument drift would be tolerated

Enclosure "C"

during a hydrogen burn than during normal operations).

Another possible difference is the criteria used to select test specimens, e.g., individual type testing for Eq versus generic testing for Es. It should also be noted that if the test condition for Eq for a LOCA can be shown to envelope the predicted test condition for a hydrogen burn then the LOCA qualification test would be sufficient to demonstrate survivability.

This requirement would apply to all BWRs and PWRs, for which construction permits were issued prior to March 28, 1979, that do not have an inerted containment atmosphere for hydrogen control. That is, plants for which there exists the possibility that substantial amounts of hydrogen can be burned in the containment will be covered by the proposed new requirement. Safety systems provided on these plants that are needed (a) to shut down the reactor and bring it to and maintain it in a safe cold shutdown condition, and (b) to prevent loss of containment integrity, must meet the "survivability" criteria in the near term and may be required to meet "qualification" criteria in the long term. Thus, for example, if a distributed igniter system is selected for controlling large amounts of hydrogen, the applicants or licensees must assure in the near term that the specified safety systems can survive and continue to perform their needed safety functions during and following hydrogen burning. In the long term the equipment may be required to meet a more stringent equipment qualification standard, considering the environmental effects of hydrogen burning. If no new hydrogen control system is required, as is likely to be the case for PWRs with large dry containments, these applicants and licensees would still have to perform analyses to: (1) Show containment structural integrity, as defined in § 50.44(c)(3)(iv) can be maintained; and (2) assure that the specified safety systems can continue to perform their needed safety functions during and following hydrogen burning and local detonations. The new criteria for certain identified essential systems are needed because the environmental pressures and temperatures associated with hydrogen burning and local detonations can be more severe than the conditions for which the equipment has been previously qualified.

Analyses [§ 50.44(c)(3)(vi)]

The proposed Interim Rule required that for all PWR and BWR plants, except the Mark I and II BWRs, design analyses must be performed for new hydrogen control measures. Many commenters indicated that the description of the design analyses was not precise enough to elicit the desired response. Furthermore, several commenters have suggested that it is inappropriate to have a regulation requiring hydrogen control design studies in view of the fact that unambiguous event descriptions and acceptance criteria are not supplied. The Commission agrees with these comments in part. As a result, the Commission intends to provide supplementary guidance concerning acceptable procedures that should be used, both for design of the hydrogen control systems per § 50.44(c)(3)(iv), for the demonstration of equipment survivability per § 50.44(c)(3)(v), and for the analysis of containment structural integrity.

The Commission is considering three different approaches concerning the supplementary guidance to be provided for performing the analyses. In all of these approaches, licensees are not restricted to the specified scenarios. If because of unique plant design features, other scenarios are known to present a greater risk than those identified by the Commission, the analyses should be based on the scenarios known to present the greatest risk. For example, if for a particular plant an intermediate break LOCA results in a greater risk than the scenarios in Table I, the licensee should base his calculations on the intermediate break LOCA scenario.

In the first approach, the Commission would identify accident sequences or scenarios which are found by probabilistic risk assessment techniques to be significant contributors to the likelihood of core degradation and thus pose a significant hydrogen threat. The licensee would then perform analyses, using these sequences, to determine the time variation of the hydrogen and steam release rates to the containment building. The analyses, which would include the failure assumptions of the different scenarios as well as the accident recovery phase and allowances for uncertainties, would provide the pressure and temperature histories to which the containment would be exposed. A list of possible accident sequences being considered under this approach is given in Table I. The scenarios include the production of substantial amounts of hydrogen as part of core-melt sequences; they were selected, based on experience and engineering judgment, because they are the more probable severe accident sequences which could be terminated short of primary vessel melt-through with available recovery techniques.

In the second approach, a base sequence would be chosen by the Commission based on its significance and characteristics from the standpoint of hydrogen threat. Key aspects of this scenario would then be parametrically varied, by the licensee, in determining the acceptability of the hydrogen control system or the containment response. This would provide a wider range than that of the selected base sequence alone. The acceptability of the analyses used in this approach would depend on the selection and range of the parameters being varied. The range must be chosen to include the effects of physically realistic degraded core accident scenarios with recovery. If licensees have determined that because of their own plant design another scenario presents a greater risk than the small break LOCA, the scenario presenting the greater risk should be chosen for parametric study. The variables and values studied should be determined on a case-by-case basis depending on the particular scenario. Table II represents a preliminary list of parameter variations that appear to provide reasonable extensions of a PWR small-break scenario (Item 1 of Table I). A corresponding BWR list has not yet been prepared.

In the third approach, the Commission would use a set of accident sequences as in Table I, and perform analyses which would define a reasonable envelope of time histories of hydrogen and steam release rates into the containment building. This envelope definition could be based on variations in the progression of different sequences and/or variations due to uncertainties within a particular sequence. The envelope of hydrogen and steam source terms to the containment would then be provided to all licensees for use in subsequent analyses. This approach would avoid the need for case-by-case sequence analyses using codes like MARCH and involving extensive iterative review of the MARCH analyses with the Commission. The intent would be for the Commission to provide hydrogen and steam source terms generic to each reactor type (BWR or PWR) and let the licensees' and NRC's ensuing attention be on the containment analysis. (The staff intends to publish for comment these generic source term analyses during the comment period for this proposed rule.)

Enclosure "C"

TABLE I.—ACCIDENT SEQUENCES LEADING TO A SIGNIFICANT HYDROGEN THREAT

PWR	1. Small LOCA with temporary loss of emergency core cooling (ECC) injection.
	2. Transient with temporary loss of all feedwater and the high pressure ECC system.
	3. Interruption of all AC electric power with failure of the auxiliary feedwater system.
BWR	4. Transient with reactor isolation and temporary failure of all coolant make-up systems.
	5. Small LOCA with temporary failure of ECC injection.
	6. Transient with failure of reactor shutdown systems and interruption of ECC systems.

TABLE II.—PARAMETRIC VARIATIONS OF A PWR SMALL-BREAK SCENARIO

Rate of H_2 release[2] (lb/min)	Timing of H_2 release	Rate of steam/enthalpy release (lb/min (millions of Btu/min))	Concurrent failures and recoveries
2	Starting at time		
10	of Uncovering of Top	600(1)	Fans.
30	of core	3,600(6)	Containment.
100	Prior to major steam release.	[1] 10,000(16)	Sprays.
1,000	Concurrent with major steam release.		All AC power.
	Following major steam release.		Recirculation.

[1] This high rate of steam release may occur for about 10 min. during ECC recovery.
[2] These rates should be assumed to be constant during the period of release and represent release from the primary system to the containment building.

The Commission particularly welcomes comments concerning which of the above approaches is preferred as well as suggestions regarding improvements or other alternatives.

The proposed rule has also been modified to clarify the types of analyses required. They can be grouped into four classes, depending upon containment design, as follows:

1. BWRs with Mark I and II type containments are required to be inerted by the companion rule on inerted containments appearing elsewhere in this issue. (See Table of Contents under NRC Rules and Regulations.) There are no further anslyses required of these plants.

2. Effective [one year after the effective date of the rule], or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later, analyses would be required for BWRs with Mark III type containments and PWRs with ice condenser type containments to demonstrate that the installed hydrogen control system is adequate and will perform its intended function in a manner that provides adequate safety margins. Analyses should also be performed to assess the effectiveness of alternative systems.

3. Effective [one year after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later, additional analyses, described under item 4, would be required for BWRs with Mark III type containments and PWRs with ice condenser type containments, to show that safe shutdown will be assured and containment structural integrity maintained during degraded core accidents.

4. Owners of all other containments would be required to perform and submit by [two years after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later: (i) Analyses to assure that during degraded core accidents containment structural integrity will be maintained: and (ii) equipment survivability analyses to assure continued containment integrity and safe shutdown capability. These degraded core accidents will be assumed to produce hydrogen releases to the containment resulting from the containment reaction of up to and including 75% of the fuel cladding surrounding the active fuel region with water for a range of time periods consistent with the accident scenarios analyzed.

The analyses required by this section serve two purposes. First, they support continued reliance on the interim requirements of this rule. Second, the results will be considered in a longer term rulemaking on degraded cores.

Paperwork Reduction Act

The proposed rule will be submitted to the Office of Management and Budget for clearance of the application requirements that may be appropriate under the Paperwork Reduction Act (Pub. L. 96–511). The SF–83 "Request for Clearance," Supporting Statement, and related documentation submitted to OMB will be placed in the NRC Public Document Room at 1717 H Street NW., Washington, D.C. 20555. The material will be available for inspection and copying for a fee.

Regulatory Flexibility Act

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this proposed rule does not fall within the purview of the Act.

Accordingly, notice is hereby given that, pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of title 5 of the United States Code, adoption of the following amendments to 10 CFR Part 50 is contemplated.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 reads as follows:

Authority: Secs. 103, 104, 161, 182, 183, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2239); secs. 201, 202, 206, 88 Stat. 1243, 1244, 1246 (42 U.S.C. 5841, 5842, 5846), unless otherwise noted. Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80–50.81 also issued under sec. 184, 68 Stat. 954, as amended: (42 U.S.C. 2234). Sections 50.100–50.102 issued under sec. 186, 68 Stat. 955: (42 U.S.C. 2236). For the purposes of sec. 223, 68 Stat. 958, as amended: (42 U.S.C. 2273), § 50.54(i) issued under sec. 161i, 68 Stat. 949: (42 U.S.C. 2201(i)), §§ 50.70, 50.71 and 50.78 issued under sec. 161o, 68 Stat. 950, as amended: (42 U.S.C. 2201(o)) and the Laws referred to in Appendices.

2. In § 50.44, paragraph (c) is amended by adding new subparagraphs (3) (iv), (v) and (vi) to read as follows:

§ 50.44 Standards for combustible gas control system in light water cooled power reactors.

* * * * *

(c) * * *

(3) * * *

(iv) Effective [one year after effective date of the rule], or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later, each boiling light-water nuclear power reactor with a Mark III type containment and each pressurized light-water nuclear power reactor with an ice condenser type containment, for which a construction permit was issued prior to March 28, 1979, shall be provided with an acceptable hydrogen control system justified by suitable programs of experiment and analysis. The hydrogen control system must be capable of handling an amount of hydrogen equivalent to that generated from the

Enclosure "C"

reaction of 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) with water, without loss of containment structural integrity (i.e., steel containments must meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. Concrete containments must meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone. These subsubarticles have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H Street NW., Washington, D.C.) If the hydrogen control system relies on post-accident inerting, the containment structure must be capable of withstanding the increased pressure (A) during the accident, where it must not exceed Service Level C Limits or the Factored Load Category (as previously specified in this paragraph) and (B) following inadvertent full inerting that may occur during normal plant operations, where it must not exceed either Service Level A Limits (for a steel containment) or the Service Load Category (for a concrete containment). Equipment required to establish and maintain safe cold shutdown and containment integrity must be designed and qualified for the environment caused by post-accident inerting. Furthermore, inadvertent full inerting during normal plant operations must not adversely effect systems and components needed for safe operation of the plant. Modest deviations from these criteria will be considered by the Commission if good cause is shown.

(v) Each light-water nuclear power reactor, for which a construction permit was issued prior to March 28, 1979, that does not rely upon an inerted atmosphere to control hydrogen inside the containment, shall be provided with systems necessary to establish and maintain safe cold shutdown and maintain containment integrity that are capable of performing their functions during and after being exposed to the environmental conditions created by the burning (or local detonation) of hydrogen. The amount of hydrogen to be considered is equivalent to that generated from the reaction of 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) with water. This requirement shall be effective as follows: for each boiling light-water nuclear power reactor with a Mark III type containment and each pressurized light-water nuclear power reactor with an ice condenser type containment, on [one year after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later; for every other light-water nuclear power reactor that must meet this requirement, on [two years after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later.

(vi) Analyses shall be performed and submitted to the Director of Nuclear Reactor Regulation for each light-water nuclear power reactor, for which a construction permit was issued prior to March 28, 1979, to evaluate the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the reaction of up to and including 75 percent of the fuel cladding surrounding the active fuel region with water) including consideration of hydrogen control measures as appropriate. Each analysis must include the period of recovery from the degraded condition. The accident scenarios to be used in the analyses must be acceptable to the NRC staff. The scope and implementation requirements for the analyses for the various types of light-water nuclear power reactors are as follows:

(A) For each boiling light-water nuclear power reactor with a Mark III type containment and each pressurized light-water nuclear power reactor with an ice condenser type containment, analyses shall be performed that justify the selection of the hydrogen control system required by § 50.44(c)(3)(iv). These analyses shall be completed and submitted by [one year after the effective date of the rule], or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later.

(B) For each light-water nuclear power reactor that does not rely upon an inerted atmosphere to control hydrogen inside the containment, analyses shall be performed to show that containment structural integrity as defined in § 50.44(c)(3)(iv) will be maintained, and systems and components necessary to establish and maintain safe cold shutdown and maintain containment integrity will be capable of performing their functions during and after being exposed to the environmental conditions created by the burning of hydrogen, including the effect of local detonations. These analyses shall be completed and submitted as follows: for each boiling light-water nuclear power reactor with a Mark III type containment and each pressurized light-water nuclear power reactor with an ice condenser type containment, by [one year after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later; for every other light-water nuclear power reactor for which these analyses are required, by [two years after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later.

Dated at Washington, D.C., this 18th day of December 1981.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 81-36558 Filed 12-22-81; 8:45 am]

BILLING CODE 7590-01-M

CIVIL AERONAUTICS BOARD

14 CFR Part 250

[EDR-436; Economic Regulations Docket No. 39932]

Denied Boarding Compensation Rules; Comprehensive Review

December 9, 1981.

AGENCY: Civil Aeronautics Board.

ACTION: Notice of Proposed Rulemaking.

SUMMARY: The CAB is initiating a comprehensive review of its oversales and denied boarding compensation rules as part of its examination of consumer protection regulations prior to sunset. The Board is seeking comment on, first, eliminating all governmental oversight in this area and, second, retaining the present rules with modifications. This rulemaking is at the Board's initiative.

DATES: Comments by: February 22, 1982; Reply comments by: March 9, 1982.

Comments and other relevant information received after this date will be considered by the Board only to the extent practicable.

Requests to be put on the Service List: January 7, 1982.

Enclosure "C"

Proposed Rules

Federal Register
Vol. 47, No. 38
Thursday, February 25, 1982

This section of the FEDERAL REGISTER contains notices to the public of the proposed issuance of rules and regulations. The purpose of these notices is to give interested persons an opportunity to participate in the rule making prior to the adoption of the final rules.

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Interim Requirements Related to Hydrogen Control; Extension of Comment Period and Editorial Corrections

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule; extension of comment period and editorial corrections.

SUMMARY: The Nuclear Regulatory Commission is extending the public comment period on its notice of proposed rulemaking, published on December 23, 1981 (46 FR 62281), for an additional 45-day period. This will provide additional time for interested members of the public to evaluate the issues raised and to develop comments on the proposed rule. The proposed rule would amend 10 CFR Part 50 to improve hydrogen control capability during and following an accident in light-water reactor facilities. The public comment period was scheduled to expire on February 22, 1982.

DATES: The new comment period expires April 8, 1982. Comments received after that date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before that date.

ADDRESS: Written comments or suggestions for consideration in connection with the proposed amendments should be submitted to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. Copies of comments received and a copy of NUREG/CR-2540, when available, may be examined in the Commission's Public Document Room at 1717 H Street NW, Washington, D.C.

FOR FURTHER INFORMATION CONTACT: Morton R. Fleishman, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone (301) 443-5981.

SUPPLEMENTARY INFORMATION: This document also corrects errors that appeared in the notice of proposed rulemaking published in the Federal Register on December 23, 1981 (46 FR 62281) as follows:

1. Table II on page 62284 is corrected to read as follows:

Table II. Parametric Variations of a PWR Small-Break Scenario

Rate of H_2 Release[1] [lb/min]	Timing of H_2 Release	Rate of Steam (Enthalpy) Release [lb/min (millions of Btu/min)][2]	Concurrent Failures & Recoveries
2	- Starting at time		
10	of uncovering of top	- 600(1)	- Fans
30	of core	- 3,600(6)	- Containment
100	- Prior to major	- 10,000(16)[3]	sprays
	steam release		- All AC power
1,000	- Concurrent with		- Recirculation
	major steam release		
	- Following major		
	steam release		

[1] These rates should be assumed to be constant during the period of release and represent release from the primary system to the containment building.

[2] The conversion from mass rate to enthalpy rate is based on 1600 Btu/lb which is believed to be appropriate for steam which is superheated by excessively hot fuel.

[3] This high rate of steam release may occur for about 10 min. during ECC recovery.

2. The third paragraph following the tables in the first column of page 62284 should read as follows:

1. BWRs with Mark I and II type containments are required to be inerted by the companion rule on inerted containments that appeared in the Federal Register on December 2, 1981 (46 FR 58484). There are no further analyses required of these plants.

On page 62283, it was indicated that the Commission would publish for comment hydrogen and steam generic source terms as part of the third approach it was considering for performing the hydrogen design analyses. A report on these source terms, NUREG/CR-2540 (BMI-2090), "A Method for the Analysis of Hydrogen and Steam Release to Containment During Degraded Core Cooling Accidents", is being issued and will be sent to those persons on the mailing list for the proposed rule. Comments on the report may be included with comments on the proposed rule.

Enclosure "D"

Dated at Washington, D.C., this 19th day of February 1982.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 82-5046 Filed 2-24-82; 8:45 am]

BILLING CODE 7590-01-M

FEDERAL HOME LOAN BANK BOARD

12 CFR Parts 531 and 563

[No. 82-105]

Transfer and Repurchase of Government Securities

February 18, 1982.

AGENCY: Federal Home Loan Bank Board.

ACTION: Proposed rule.

SUMMARY: The Board proposes to amend its regulations concerning retail repurchase agreements to confirm and expand significant consumer protections, including the prohibition against the sale of retail repurchase agreements by insured institutions which do not meet the Board's net-worth requirement, the requirements that retail repurchase agreement purchasers be given a perfected security interest in the security or securities underlying retail repurchase agreements, that the securities underlying retail repurchase agreements be marked-to-market on a monthly basis, and that prospective retail repurchase agreement purchasers be provided with offering documents which contain full and accurate disclosure of all material information regarding the retail repurchase agreement and the issuing institution. In addition, the Board proposes to delete the current regulatory prohibition against the automatic renewal of retail repurchase agreements.

DATE: Comments must be received by: March 29, 1982.

ADDRESS: Please send comments to Information Services, Office of General Counsel, Federal Home Loan Bank Board, 1700 G Street, NW., Washington, D.C., 20552. Comments will be available for public inspection at this address.

FOR FURTHER INFORMATION CONTACT: Donna K. Ralston (202-377-6417) Office of General Counsel, Federal Home Loan Bank Board, 1700 G Street, NW., Washington, D.C. 20552.

SUPPLEMENTARY INFORMATION:

Background

On August 2, 1979, the Board amended the Federal Home Loan Bank System Regulations to provide in § 531.12 (12 CFR 531.12; 44 FR 46445, August 8, 1979) that a member of a Federal Home Loan Bank may issue to the public "obligations * * * evidencing an indebtedness arising from the transfer of direct obligations of, or obligations that are fully guaranteed as to principal and interest by, the United States or any agency thereof that [the] member institution is obligated to repurchase", provided that the obligations, commonly referred to as retail repurchase agreements, are issued in denominations less than $100,000, have a maturity less than 90 days, are not subject to automatic renewal or extension, and have the following legend:

> This obligation is not a savings account or deposit and is not insured by the Federal Savings and Loan Insurance Corporation.

In order to permit member institutions to sell retail repurchase agreements at their offices, the Board on February 13, 1981, amended § 563.8(f) of the Insurance Regulations (12 CFR 563.8(f); 46 FR 13982, February 24, 1981) to exempt retail repurchase agreements from the minimum denomination rule applicable to outside borrowings.

To provide guidance to issuing member institutions, on September 9, 1981, the Board's staff issued R Memorandum No. 51a, which set forth the staff's views regarding the requirements imposed by § 531.12 and other regulations of the Board on retail repurchase agreements. The Board has found that the staff views expressed in R Memorandum No. 51a constitute a reasonable interpretation of the applicable regulations and now proposes to formally adopt several of those interpretations in regulatory form. Moreover, the Board believes that the confirmation and expansion in its regulations of certain consumer protections will ensure that insured institutions will be able to offer and sell to their customers superior consumer investments that will combine competitive market rates and significant investor security.

Because retail repurchase agreements are borrowings, the Board believes that it would be appropriate to redesignate § 531.12 as § 563.8-4 of the Board's Insurance Regulations, and to amend the regulation to expressly establish the requirements that insured institutions must meet in connection with the issuance of retail repurchase agreements. In addition, because the proposed regulations would establish significant consumer protections, the Board proposes to remove the prohibition against the automatic renewal of retail repurchase agreements. This will substantially lessen administrative costs to issuing institutions and, therefore, enable issuing institutions to offer and consumers to receive higher rates of return. Also, it will give issuing institutions greater flexibility in developing competitive retail repurchase agreement programs.

Proposed Regulation

Proposed § 563.8-4 provides the following:

1. The interest of the purchaser in the security or securities underlying a retail repurchase agreement shall constitute a perfected security interest under applicable state law.
2. The market value of the security or securities underlying a retail repurchase agreement shall be at least equal to the principal amount of the issuing institution's obligation as of the date of the original issuance of the retail repurchase agreement and as of a date certain in each succeeding month of the original or renewed term of the repurchase agreement.
3. An institution issuing retail repurchase agreements shall provide to each prospective purchaser an offering document which shall contain full and accurate disclosure of all material information regarding the retail repurchase agreement and the issuing institution. Any significant change in any of the material representations set forth in the offering document shall be reflected in a revised offering document which shall be provided to retail repurchase agreement purchasers before any renewal of a retail repurchase agreement may be effected.
4. An institution which does not meet the net worth required under § 563.13(b) of institutions that have reached the twentieth anniversary of insurance of accounts shall be prohibited from issuing retail repurchase agreements. An institution that fails to meet the net-worth requirement at a time when it has retail repurchase agreements outstanding shall be prohibited from renewing its outstanding retail repurchase agreements.
5. An institution issuing retail repurchase agreements shall not use in its advertisements or offering documents the terms "guaranteed", "no risk", "account" "deposit", "withdrawal" or any other terms that imply that the retail repurchase agreement is insured or guaranteed by the United States government or any agency of the United States government, or the term "fund", or any other terms that imply that a retail repurchase agreement constitutes an interest in an investment company. In addition, an institution issuing retail repurchase agreements shall state in its

Enclosure "D"

COMMENT LETTERS FOR HYDROGEN RULE

Letter No.	Date	Organization	Commenter	No. of Comments
1	1/25/82	Commonwealth Edison	L. O. DelGeorge	6
2	2/8/82		S. L. Hiatt	5
3	2/11/82	Westinghouse Electric	E. P. Rahe, Jr.	12
4	2/19/82	Stone & Webster	R. B. Bradbury	18
5	2/22/82	Power Authority of N.Y.	J. P. Bayne	7
6	2/16/82	Alabama Power	F. L. Clayton	4
7	2/23/82	C-E Power Systems	A. E. Scherer	6
8	2/18/82		J. D. Parkyn	5
9	2/23/82	Florida Power	D. G. Mardis	4
10	2/26/82	Bechtel Power Corporation	A. L. Cahn	10
11	3/1/82	Houston Lighting & Power	C. G. Robertson	5
12	3/25/82	Commonwealth Edison	L. O. DelGeorge	2
13	3/31/82	Industry Degraded Core Rulemaking (IDCOR) Program	C. Reed	8
14	3/31/82	Tennessee Valley Authority	L. M. Mills	10
15	4/6/82	Washington Public Power	F. D. Bouchey	7
16	4/6/82	General Electric	G. G. Sherwood	8
17	4/8/82	Northeast Utilities	W. G. Counsil	6
18	4/6/82	Wisconsin Electric	C. W. Fay	12
22		Missississpi Power & Light	J. P. McGaughy, Jr.	4
23	4/8/82	Hydrogen Control Owners Group	J. D. Richardson	16
24	4/8/82	Portland General Electric	B. D. Withers	5
25	4/8/82	Nuclear Utility Group on Equipment Qualification	N. S. Reynolds	4
26	4/9/82	Yankee Atomic Electric	D. W. Edwards	3
27	4/8/82	Gulf States Utilities	J. E. Booker	9
28	4/8/82	Duke Power	W. L. Porter	10
29	4/5/82	Texas Utilities Generating Co.	R. J. Gary	2
30	4/6/82	GPU Nuclear	J. R. Thorpe	8
33	4/12/82	Louisiana Power & Light	L. V. Maurin	6
			Total	202

Enclosure "E-1"

TALLY OF COMMENT LETTERS

Twenty-eight applicable comments have been received with the sources distributed as follows:

Nuclear steam system suppliers	3
Utilities	18
Architect/engineer firms	2
Industrial associations	3
Individuals	2
	28

Note:

1. Comment 19 identical to Comment 16
2. Comment 20 applied to a different final rule (46 FR 58484)
3. Comment 21 identical to Comment 15
4. Comment 31 applied to a different final rule (46 FR 58484)
5. Comment 32 identical to Comment 17

Enclosure "E-2"

LIST OF COMMENTS

1. Commonwealth Edison - Utility

Comment 1: Improvements in hydrogen control for small non-inerted containments is warranted.

Comment 2: Hydrogen survivability considerations for inerted BWRs and large, dry PWRs should be deferred to the long term degraded core rulemaking.

Comment 3: The 75% metal-water reaction is reasonable but plants should be able to analyze accident sequences to see if a combustible mixture can be formed.

Comment 4: The added conservatism associated with the Eq approach is not warranted for a low probability event and no need for the conservatism has been demonstrated.

Comment 5: The survivability rule may be counterproductive to safety by causing replacement of reliable equipment with equipment of a new design with less operating history.

Comment 6: The first approach, using recommended accident sequences for the analyses, is preferred. Flexibility in the selection of the accident sequences should be permitted.

2. Susan L. Hiatt - Individual

Comment 1: It is unrealistic to require analyses without giving any criteria for their evaluation.

Comment 2: The Commission appears to be soliciting suggestions from the licensees as to what the requirements should be. The licensees should not be consulted.

Comment 3: The analysis is only intended to justify the hydrogen control system already installed; not to install the most effective one. Analyses should be required before the plant is constructed.

Comment 4: Not requiring the analyses until the plant exceeds 5 percent of rated power removes the issue from the public hearing.

Enclosure "E-3"

Comment 5: The combustible gas requirements should be as specific as the ECCS criteria. Until such regulations are promulgated, the existing § 50.44(c), requiring containment inerting, should be enforced.

3. Westinghouse Electric - Nuclear Steam System Supplier

Comment 1: The rule does not give credit for all the improvements made since the TMI accident.

Comment 2: The order of the rule should be changed with the analysis requirement coming first.

Comment 3: The 75 percent clad reaction is too large compared to what happened at TMI and based on analysis results for a recovered degraded core event.

Comment 4: The arbitrary assumption of a 75 percent clad reaction can lead to problems when combined with accident sequences. A more mechanistic approach should be used.

Comment 5: The first approach, by specifying sequences, is most appropriate since the hydrogen generation rules will be plant specific. Low probability sequences should not be considered.

Comment 6: Transients with failure of all containment safeguards should not be included.

Comment 7: In Table II the suggested upper limit on the hydrogen production rate during a small LOCA (1000 lb/min) is unrealistic. It would be less than 100 lb/min due to break flow being choked.

Comment 8: It is inappropriate to require consideration of local detonations in demonstrating equipment survivability since the probability of occurrence of a local detonation is extremely small.

Comment 9: The issue of equipment qualification for a hydrogen burn should be kept separate from equipment qualification for design basis events to avoid additional complexity and inconsistencies in implementation of the two.

Comment 10: A two-phased approach to equipment qualification criteria will only add to the financial impact. The survivability concept is logical and should be issued in final form.

Comment 11: The survivability criteria should apply only to systems necessary for "safe shutdown" rather than "safe cold shutdown." "Cold shutdown" would require a new design basis for many plants.

Comment 12: The proposed containment structural integrity limits when coupled with the suggested accident sequences will likely result in "calculated" containment failures. The criteria are much too restrictive and go beyond merely addressing hydrogen control for a "TMI-like accident." A realistic value of structural capability should be allowed along with the use of actual material properties (rather than minimums) and realistic analyses (i.e., no concurrent multiple failures).

4. Stone and Webster - Architect/Engineer Firm

Comment 1: The interim rule should only be temporary pending completion of the severe accident rulemaking and should only address basic concerns such as containment failure and fission product release from a postulated hydrogen burn.

Comment 2: Analysis should only be required for a realistic source. If ultimate strengths are not exceeded, no further analysis should be done. Implementation of new design changes should await the severe accident rulemaking.

Comment 3: Only a date for submitting design analyses schedules should be required. The actual date for analyses submitted should be left on a case-by-case basis.

Comment 4: The criteria for whether or not a hydrogen control system should be added, should include an analytical demonstration, such as a PRA, that there would be a net safety improvement by its addition.

Comment 5: Is the 75 percent limit reasonable for BWRs? What about other potential hydrogen generating reactions such as with iron and other metals? What about credit for ECCS performance?

Comment 6: What is the basis for saying that control methods not involving burning provide protection for a wider range of accidents than those that involve burning? Why are deliberate ignition systems deemed acceptable for interim approval?

Comment 7: Equipment qualification should not be part of the hydrogen control rulemaking but should be addressed separately.

Comment 8: A two-step approach to equipment qualification is not practical since it makes no sense to replace or requalify equipment based on a "survivability" standard if it would have to be requalified to a stricter standard in the near future.

Comment 9: Why have different implementation dates for Mark IIIs and ice condensers than for PWRs?

Comment 10: The accident scenarios referenced appear to relate to LOCA scenarios which may not be the same as the worst hydrogen scenarios.

Comment 11: Will a review be required to identify scenarios having a greater risk than those specified or need they be addressed only if already identified elsewhere?

Comment 12: If because of unique plant design features the likelihood of a given accident sequence is small, it should not need to be analyzed.

Comment 13: Table II is confusing regarding its implementation.

Comment 14: The third approach is the best as it would put all plants on an equal basis and provide a better comparison of containment responses.

Comment 15: If analyses show that containment integrity will not be maintained, plant modifications should not be required without an integrated evaluation considering PRA, safety goals and severe accident rulemaking.

Comment 16: Mark I/II reactors should be allowed some other form of hydrogen control besides preinerting.

Comment 17: "Maintaining containment structural integrity" is not as important a concern as "mitigating radiological releases which could jeopardize public health and safety." The rules should be revise to reflect this comment.

Comment 18: Comparative analyses of systems should not be required, only a demonstration that the chosen system works.

5. Power Authority of the State of New York - Utility

Comment 1: The CRGR should review the rule to ensure that an integrated assessment and a cost/benefit analysis is performed to determine the need for the rule.

Comment 2: The rule would impose significant analytical and equipment installation requirements with no assurance that safety will be improved.

Comment 3: No dates should be set until the supplementary guidance is available. Furthermore, the dates for completion of analyses and equipment installation should not coincide to ensure sufficient time for mechanical work. (Suggested new wording in letter.)

Comment 4: Finite element stress analysis of the containment shell and fracture mechanics analysis of the steel liner should be allowed to verify containment integrity.

Comment 5: Required analytical tools that are approved and checked out are not currently available.

Comment 6: Credit should be given for facility modifications that prevent a degraded core accident and thus avoids hydrogen production.

Comment 7: Hydrogen control questions should be deferred to the severe accident rulemaking since improvements may be made which prevent a DCA.

6. Alabama Power Company - Utility

Comment 1: The proposed rule should only address non-inerted, small-volume, low-pressure containments and not require analyses and backfitting for other containments that would provide only marginal, at best, improvements in safety.

Comment 2: The 75 percent metal-water reaction is not supported by research information. Furthermore, a DCA is significantly less likely now than it was at the time the 5 percent metal-water reaction criterion was established.

Comment 3: The requirement for equipment qualification for systems necessary for safe cold shutdown is a significant backfit. The current licensing criterion in many cases is for hot shutdown capability not cold shutdown capability. The issue of cold vs. hot shutdown should be deferred to a separate rulemaking since it involves a significant backfit and is only marginally related to hydrogen control.

Comment 4: Environmental qualification has recently been fully addressed in response to the Commission and to reevaluate equipment inside containment for a hydrogen deflagration environment is not justified since it has not been demonstrated that safety would be improved.

7. C-E Power Systems - Nuclear Steam System Supplier

Comment 1: Equipment survivability should not be required until a safety goal has been established and a determination made of the degree to which degraded cores should be considered in safety regulation. A cost/benefit analysis should be done.

Comment 2: There does not appear to be coordination with the proposed rule on qualification of electrical equipment and there will be overlapping of requirements.

Comment 3: It is premature to select 75 percent for the metal-water reaction since it neglects the improvements made since the TMI accident as well as the results of current research studies that indicate that there is a natural phenomena which tends to limit hydrogen generation.

Comment 4: The two-step procedure for equipment qualification is not justified in view of the lack of indication that the level of safey needs to be increased.

Comment 5: Imposing extra margins for equipment qualification to account for uncertainties in a low probability event will not increase safety and may even be counter-productive to safety by precluding the use of otherwise reliable equipment.

Comment 6: When and if supplementary guidance is provided, it whould be in the form of acceptance criteria related to an overall safety goal and should allow flexibility with regard to the approach used provided that a certain level of safety is achieved.

8. John Parkyn - Individual

Comment 1: In view of the fact that the TMI accident showed that the containment did not fail and that vital equipment continued to function after the detonation, it is not justified to expand the scope of an existing environmental qualification program that is already of questionable value.

Comment 2: The environmental qualification effort should be delayed until after the extent of core damage at TMI is ascertained.

Comment 3: The environmental qualification program is not needed because the DCA is such a low frequency event.

Comment 4: If only a LOCA can be turned into a DCA by human error, then events which break containment directly are of greater concern than hydrogen generating events.

Comment 5: The rule should not apply to plants that have stainless steel clad fuel elements since they do not have a hydrogen production problem.

9. Florida Power Corporation - Utility

Comment 1: It is inappropriate to require qualification to environmental conditions that are yet to be determined.

Comment 2: The two years requirement on equipment survivability verification should account for prior analyses as well as analyses required by § 50.44(c)(3)(vi).

Comment 3: It is reasonable to require the determination of the survivability of installed equipment. It will allow a cost/benefit analysis for equipment replacement decisions.

Comment 4: Analyses should not be required to include a certain amount of hydrogen generated but should include a determination of the amount of hydrogen generated during a worst case accident thus producing a conservative answer to the question of equipment survivability.

10. Bechtel Power Corporation - Architect/engineer Firm

Comment 1: It is not indicated that the rule only includes interim requirements. When and how will it be rescinded?

Comment 2: The 75 percent metal-water reaction is not justified and may impose overly restrictive requirements. It should be used as a default value but licensees should be permitted to use other values if justified by research, scenario definition and detailed analysis.

Comment 3: The implementation schedules may be impossible to meet in view of the fact that the survivability criteria have not yet been determined and the available testing facilities are committed to NUREG-0588 qualification testing. Realistic implementation schedules should be established on a plant by plant basis.

Comment 4: The criteria for containment integrity should not be limited to the provisions of the ASME Boiler and Pressure Vessel Code but should permit licensees to demonstrate containment integrity using mutually agreed upon methods. The ASME Code can be cited as an example of an acceptable means for the demonstration.

Comment 5: The proposed limits for concrete containments are overly restrictive and should be increased by a factor of 1.5 since the containments are designed to withstand pressure that is 1.5 times the accident pressure.

Comment 6: Comparative analyses of different hydrogen control systems should not be required since it is not required for other systems. Satisfaction of specific criteria should be sufficient.

Comment 7: The concept of a two-step approach to equipment survivability and qualification sounds reasonable but it was not described in sufficient detail. The criteria for survivability should take into account TMI and other operating experience.

Comment 8: The third approach for supplemental guidance on analyses appears preferable since it would minimize the amount of repetitive analysis and review required. It is essential that sufficient industry and PRA type input be utilized in the scenario definition.

Comment 9: With regard to local detonations, provisions should be made to allow arguments as to why detonations could not occur, or alternatively, detonation parameters should be provided.

Comment 10: While plants must be brought to a safe cold shutdown, the rule should not impose the use of safety related equipment to accomplish it particularly for plants whose licensing basis only requires achieving a safe hot shutdown.

11. Houston Lighting & Power - Utility

Comment 1: The TMI accident probably had close to the maximum metal-water reaction that could occur in an accident in which the containment threat is limited to the combustion of hydrogen. Furthermore, the upgrades required by NRC make a DCA less likely. The 75 percent metal-water reaction is thus not justified.

Comment 2: The first approach for guidance on analyses appears most appropriate but a probability threshold should be established to ensure that significant scenarios are identified for each plant.

Comment 3: The requirement for consideration of local detonations for equipment survivability should be justified since it is not clear they can occur in nuclear plants.

Comment 4: Hydrogen burn should not be used as part of equipment qualification since it represents a significant extension of the types of events encompassed by equipment qualification. The "survivability" concept is much more appropriate but should not be limited to only an interim period.

Comment 5: A realistic value of containment structural integrity should be used rather than defining it in terms of service level C and the factored load category.

12. Commonwealth Edison - Utility

Comment 1: Table II should be revised in light of the NRC sponsored analyses presented in NUREG/CR-2540. The peak hydrogen and steam release rates are too high.

Comment 2: The 75 percent metal-water reaction assumption is not realistic in light of the data presented in NUREG/CR-2540 since is would seem to imply that ECCS is restored within only a 10 minute window out of a time span of over 4 hours from onset of LOCA to failure.

13. Industry Degraded Core Rulemaking Program - Industrial Association

Comment 1: The Commission should state in a policy pronouncement that the Interim Hydrogen Rule in combination with the ongoing generic rulemaking on severe accidents precludes consideration of generic severe accident issues from individual plant dockets.

Comment 2: Implementation of the proposed rule should be delayed pending the outcome of the Severe Accident Rulemaking.

Comment 3: The proposed rule ignores the post TMI improvements that have been made and which reduce the likelihood of a DCA.

Comment 4: There is not sufficient safety urgency to warrant issuance of the proposed rule; the requirements go beyond the framework originally envisioned for an "interim rule."

Comment 5: Hydrogen generation is only one of several technical issues that need to be resolved for accidents beyond the DBA. It should be treated in the Severe Accident Rulemaking rather than a piecemeal approach.

Comment 6: Delay of the proposed rule will permit the completion of major research programs in the hydrogen area; which will reduce technical uncertainties and provide a better technical basis for the rule.

Comment 7: Delay will permit completion of development of a new accident analysis program (early 1983) which would be used to perform the required analyses.

Comment 8: The cost of the rule to industry has not been adequately considered. It is estimated that the survivability analysis would cost between $250K and $600K per unit for a total cost for 100 units of about $35M - $50M.

14. Tennessee Valley Authority - Utility

Comment 1: The issue of hydrogen control should be considered in the context of overall plant risk from DCA's.

Comment 2: The term "certain important safety systems" in the summary should be revised to read "certain systems important to safety," to be consistent with other NRC terminology.

Comment 3: A physically more reasonable maximum clad reaction fraction would be 30-40 percent rather than 75 percent. The parameter of greater importance, however, is the hydrogen release rate rather than the magnitude.

Comment 4: The requirement that the operation of the hydrogen control system not further aggravate an accident or endanger the plant during normal operations would seem to eliminate the post accident inerting systems.

Comment 5: It is not clear why the Commission believes that hydrogen control methods that do not involve burning would provide protection for a wider spectrum of accidents than those that involve burning particularly if all ramifications are considered.

Comment 6: Since the consideration of severe accidents goes beyond the design basis for existing plants, the only requirement for systems that must function during a hydrogen burn should be that they "survive" and continue to be able to perform. The two-step approach is unnecessary and proof of survivability is adequate for extensions beyond the design basis.

Comment 7: Since maintenance of core cooling is mainly dependent on active systems outside containment, a rigorous burn "qualification" program on essential equipment inside equipment would have little effect on reducing the likelihood of a DCA or recovering from such an event.

Comment 8: As an alternative to the consideration of local detonations, a demonstration should be permitted to show that they are unlikely.

Comment 9: The first analysis approach of specifying a small number of significant scenarios appears to be reasonable.

Comment 10: The second analysis approach may also be reasonable except that the range of parametric variation suggested in Table 2 is unrealistic. Thus, while the base scenario may be reasonable, the introduction of arbitrary additional equipment failures represents a different scenario with a much lower occurrence probability and thus having a lower risk contribution. Analysis of events beyond the design basis should be performed as realistically as possible.

15. Washington Public Power Supply System - Utility

Comment 1: The hydrogen rule should be delayed pending completion of the severe accident rulemaking since the priority technical issues related to hydrogen have already been addressed and the severe accident rulemaking will address this and other technical issues in a more comprehensive framework.

Comment 2: The rule represents premature judgment in requiring mitigation for degraded core scenarios and extends the design bases to include degraded cores without sufficient technical justification.

Comment 3: The rule should be modified so that if it is shown that a method for controlling hydrogen concentration so as to prevent a hydrogen burn is supplied, then the equipment survivability criteria does not have to be demonstrated.

Comment 4: The rule is tantamount to requiring utilities to have the capability to mitigate Class 9 accidents, an extreme shift in the design basis of current plants, without the benefit of formal rulemaking. The option for utilities to make cost effective choices between prevention and mitigation is lost and the IDCOR effort is subverted.

Comment 5: The requirement for equipment survivability represents an open-ended rachet for equipment qualification in view of the ambiguity involved. For example, the Es models have no experimental basis and hence, no criteria for judging their acceptability; for Es, the tests would have to be redone whenever new analyses were done and it is not clear that test facilities could be found to match the environments; no criterion is provided as to what constitutes acceptability in "perform its function"; since Es could be treated on a case-by-case basis everyone could be qualifying to separate performance standards.

Comment 6: The three suggested approaches still do not provide the "unambiguous event descriptions and acceptance criteria" that are needed. No acceptance criteria are proposed and the event descriptions are still arbitrary and ambiguous. A safety goal should be provided and the utilities permitted to use their perogatives to achieve it.

Comment 7: It is inappropriate to require design studies for large amounts of hydrogen until an appropriate level of release has been determined in the severe accident rulemaking.

16. General Electric - Nuclear Steam System Supplier

Comment 1: The requirement for hydrogen control systems for Mark III BWRs should be revised to permit a demonstration that additional hydrogen control systems are not necessary as a result of design capabilities that prevent hydrogen generation or limit its impact. The decision criteria should include consideration of the probability and consequences of hydrogen generation.

Comment 2: Because some nuclear plants employ a multi-building, multi-barrier design for the containment, the loss of containment structural integrity would not necessarily result in excessive radiation dose to the public. The rule should refer to loss of containment "function" rather than containment "integrity."

Comment 3: The 75% metal-water reaction is unrealistically high and is inconsistent with the desire to set a limit in which the threat to containment is limited to that of hydrogen combustion. The metal-water reaction should be defined by applicant performed analyses using realistic accident scenarios.

Comment 4: The requirement imposed on post-accident inerting systems, that in case of inadvertent full inerting the containment structural stresses not exceed Service Level A, is unnecessarily conservative. In view of the time required for full inerting and operator intervention inadvertent full inerting is a low probability event. The requirement should only be that containment function and assurance of safe plant shutdown be maintained in the event of inadvertent full inerting.

Comment 5: A statement should be added to the Supplementary Information indicating that the issues are sufficiently resolved to warrant interim approval of a deliberate ignition system for Mark III BWR plants.

Comment 6: Survivability criteria (Es) and qualification requirements (Eq) should be defined prior to implementation of the proposed amendments and issued for public comment before being made effective.

Comment 7: The scope of the required analysis should be expanded to permit analyses that demonstrate that additional hydrogen control systems are not needed.

Comment 8: An approach similar to the proposed first approach is recommended except that realistic accident scenarios should be defined using PRA techniques and they should be analyzed using best estimates.

17. Northwest Utilities - Utility

Comment 1: The proposed amendments on survivability of equipment and containment and the associated analyses should deferred until ongoing research programs are completed so as to provide a technical basis for the amendments.

Comment 2: Hydrogen issues are only a portion of the concerns associated with degraded cores. The proposed amendments should be deferred to the severe accident rulemaking.

Comment 3: A Regulatory Impact Analysis should be prepared and issued for comment before implementation of the proposed amendments.

Comment 4: No basis is provided for changing the licensing basis of operating plants from hot shutdown to cold shutdown. The cold shutdown issue should be considered completely and independently of the hydrogen issue.

Comment 5: The implementation schedule should not require completion of the proposed survivability program prior to completion of the equipment qualification program for DBAs. It should also coincide with refueling outages and allow time for design and implementation. Finally, it should be scheduled to follow the completion of the analysis required by § 50.44 (c)(3)(vi)(B).

Comment 6: The proposed amendment makes no mention of a two-step approach for qualification of essential equipment and is ambiguous. A one-step approach would be preferable.

18. Wisconsin Electric - Utility

Comment 1: The proposed amendments are not needed because conservative analyses and experiments demonstrate that there is a low probability of significant hydrogen generation in DBAs and an even lower probability in accidents beyond the design basis. This was reduced further by safety upgrades following the TMI-2 accident.

Comment 2: Preliminary best-estimate analyses of TMI-2 show that a hydrogen burn following a 100 percent metal-water reaction will not result in a loss of containment structural integrity or in a detonation.

Comment 3: Preliminary testing for EPRI of typical safety-related electrical equipment under hydrogen burn conditions indicate that they can survive a hydrogen burn.

Comment 4: Since there is no immediate safety need for the proposed rules, a cost-benefit analyses would show that the substantial burden on licensees with large containments is not warranted. Estimates for doing the containment analyses are $1-2 million and for equipment survivability testing are $0.5 million.

Comment 5: The proposed rule should be deferred to the Severe Accident Rulemaking and its need should be demonstrated by PRA.

Comment 6: The proposed rule should be delayed until ongoing research programs on hydrogen burning and equipment survivability are sufficiently completed to provide a technical basis for the rule.

Comment 7: Since the licensing basis of most operating plants is to achieve and maintain hot safe shutdown conditions following accidents, the proposed requirement should be modified by deleting the word "cold."

Comment 8: The 75% metal-water reaction is not technically justified and a 50% value appears to be more reasonable.

Comment 9: The accident scenarios for the analyses are not sufficiently defined or justified. A mechanistic time rate of hydrogen release should not be required if the hydrogen is assumed to start at the maximum hydrogen concentration.

Comment 10: Equipment survivability analyses should not be required to include detonation considerations since studies indicate they will not occur in large dry containments.

Comment 11: The implementation schedules are unrealistic. The equipment survivability analyses should commence after the containment analyses are completed which should take 2 years at a minimum. An estimated 2 to 5 years would be needed to meet the provisions of the rule.

Comment 12: The part of the rule which states that "the accident scenarios to be used in the analyses must be acceptable to the NRC Staff" should be deleted since it gives the NRC authority to arbitrarily change the rule. General accident scenarios should be specified, based on severe accident rulemaking and PRA studies, that also allow flexibility for plant-specific designs.

22. Mississippi Power and Light - Utility

Comment 1: The substantial quantity of information provided by industry in support of the distributed ignition system has been largely ignored in development of the rule.

Comment 2: The probability of scenarios leading to significant hydrogen generation is so small that the interim requirements are not needed.

Comment 3: Interim rules should be delayed pending completion of the ongoing research program which may demonstrate that the interim rule would provide a negligible increase in plant safety.

Comment 4: The industry cannot respond to a plethora of interim requirements and also support the severe accident rulemaking.

23. Hydrogen Control Owners Group - Industrial Association

Comment 1: Issuance of the interim rules for extremely low probability events is not advisable and should be considered in the context of the severe accident rulemaking and the safety goals.

Comment 2: Action on the rule should be deferred pending completion of the ongoing major research program which will be completed in the near future.

Comment 3: The rule makes no mention of when the interim requirements will be replaced with final requirements.

Comment 4: The consideration of large scale hydrogen releases is contradicted by NRC sponsored PRA studies that showed that the risk of containment failure due to hydrogen combustion is small compared to other risk contributors.

Comment 5: The concern over large amounts of hydrogen ignores the plethora of improvements, mandated by NRC since the TMI-2 accident, that substantially reduce the probability of degraded core accidents.

Comment 6: The discussion of the rule should be revised to indicate that the acceptability of the hydrogen control system will be assessed based on generic, rather than acceptable programs of experiment and analysis, since plant specific experiments are not justified.

Comment 7: The analyses in support of the hydrogen control system should be limited to establishing the adequacy of the selected design and it should not be required to include comparative analyses of alternative systems that may be used in system selection.

Industry has already submitted evaluations of alternate concepts to the NRC and, in the IDCOR program, will be preparing additional comparative analyses. Alternate concept studies have never been previously required for rulemakings and would represent a wasteful and inefficient allocation of industry and NRC resources.

Comment 8: The text should be modified to indicate that the Mark III owners are seriously considering a distributed igniter system for Mark III containments.

Comment 9: Because of the BWR design, which operates normally with a large steam fraction in the core, serious fuel damage and large hydrogen releases are unlikely. Studies have indicated that the maximum metal-water reaction for a BWR/6 prior to core slump is less than 12.5%. A metal-water reaction of less than 75% should be permitted if justified by analysis of realistic, mechanistic accident scenarios.

Comment 10: There is no technical justification for the statement "that control methods that do not involve burning provide protection for a wider spectrum of accidents then do those that involve burning."

Comment 11: Sufficient information has been submitted for Mark III BWRs to give interim approval of deliberate ignition systems for Mark IIIs as well as for ice condensers.

Comment 12: A two-step equipment survivability and qualification program is not warranted in view of the low probability of a DCA. A conservative analysis demonstrating survivability, with no consideration of local detonations, is all that is needed.

Comment 13: None of the three suggested approachs on the guidance for the analyses are desirable. The language is too vague permitting the possible scope of the analyses to be of unmanageable magnitude. Furthermore, the scenarios suggested are grossly conservative and are not realistic enough to assess containment response or hydrogen control effectiveness. It should be left to the applicant or licensee to choose the most probable accident scenarios and hydrogen release rates rather than for the NRC to establish arbitrary and overly conservative criteria.

Comment 14: The issue of hydrogen control should be considered in the broader context of safety goals and risk reduction. If it can be shown that hydrogen combustion only causes a slight increase in risk then additional analyses and hydrogen control is not warranted.

Comment 15: The description of containment integrity should not include the detailed ASME criteria. Instead, a range of alternative means should be permitted to demonstrate compliance; with the ASME criteria used only for illustrative purposes.

Comment 16: The implementation schedule for the submission of the required analysis is unreaslitic and cannot be met. It should be modified to avoid the necessity for numerous schedule extensions.

24. Portland General Electric - Utility

Comment 1: Since the basis for most operating plants is safe shutdown, the word "cold" should be deleted from the phrase "safe cold shutdown."

Comment 2: The 75% clad reaction is unsubstantiated and does not allow credit for the post-TMI modifications that are intended to prevent a DCA. Analyses that consider preventive measures should be used to establish a realistic cladding reaction percentage.

Comment 3: The two-step approach for qualification of equipment is not clearly defined and may involve an undue financial burden due to repeated testing of equipment.

Comment 4: Either of the first two methods for proceeding with the analyses are acceptable if preventive measures are permitted to be considered.

Comment 5: The proposed rule should be clarified with regard to the sequence of performing the analyses and providing the necessary systems. The schedule should be adjusted so that the required equipment should be provided 2 years after the analyses are completed.

25. Nuclear Utility Group on Equipment Qualification - Industrial Association

Comment 1: The proposed rule is premature, is being promulgated with no supporting technical basis, and appears to lack the proper review of its need by senior management and the Commissioners.

Comment 2: No technical justification has been presented to support the position that the temperatures and pressures associated with hydrogen burning and local detonation can be more severe than the conditions for which the equipment has been previously qualified.

Comment 3: Based on analyses and experiments of technical experts and the NRC staff, developed in support of licensing hearings, the essential equipment can survive hydrogen burning.

Comment 4: Supporting justification is required since the proposed rule is (1) interim in nature, (2) is subject to ongoing research, and (3) addresses a very remote accident beyond the design basis.

26. Yankee Atomic Electric - Utility

Comment 1: The proposed rule should be delayed pending completion of the IDCOR program related to severe accidents.

Comment 2: In view of the significant risk reduction steps taken since the TMI-2 accident, there is no urgency for the proposed rule.

Comment 3: The metal-water reaction should be established based on the results and codes developed in the IDCOR program.

27. Gulf States Utilities - Utility

Comment 1: The proposed rule should be considered in light of the broader severe accident rulemaking and the need for a hydrogen control system evaluated in the context of the long-term safety goals.

Comment 2: The implementation of the TMI Action Plan requirements has substantially reduced the probability of a DCA and credit should be allowed for the modifications.

Comment 3: There should be no requirement for a comparative analysis of alternate hydrogen control systems. Criteria should be specified and as long as the system meets the criteria it should be acceptable.

Comment 4: The 75% metal-water reaction is excessive in light of recent studies that core slump would occur before 35-40%. A mechanistic approach should be permitted to establish a realistic maximum value.

Comment 5: The containment structural integrity limits are too restrictive and actual material properties should be permitted. A realistic criteria of functional capability should be used.

Comment 6: The implementation schedules are unrealistic and should be modified.

Comment 7: A distributed ignition system should be considered generally acceptable for BWR Mark IIIs as well as for ice condenser plants.

Comment 8: The two step approach to equipment qualification is not warranted. An equipment survivability requirement is appropriate but it should be permitted to be demonstrated by analysis and should be separated from qualification for DBAs. Local detonations should not have to be considered in view of the low probability for its occurrence.

Comment 9: The first approach is preferred for the analyses, however, sequences that have a lower probability then that defined by the safety goal should not have to be considered. ATWS should not be included since the ATWS rule will ensure that this is a low probability event.

28. Duke Power - Utility

Comment 1: A post accident hydrogen control rule should not be promulgated now but should only be considered after the Severe Accident Rulemaking is complete and a safety goal is established.

Comment 2: The degree of cladding oxidation should be consistent with the accident sequence analyzed and not for the most severe sequence. The 75% limit is not consistent with the existing data.

Comment 3: It is unclear whether the version of the ASME Code referenced is the Summer of 1980 Code or the Code of Record.

Comment 4: Code limitations should not be applied to beyond DBAs. A realistic limit load analysis should be allowed to assure containment structural integrity.

Comment 5: Recent EPRI tests and recent studies, reports, and analyses, strongly indicate that the proposed survivability requirements are not needed. They cannot be justified from either a risk reduction or cost-benefit standpoint.

Comment 6: Since the design basis of most nuclear stations is safe hot shutdown under DBA conditions it is inappropriate to require safe cold shutdown.

Comment 7: Licensees should have the option of demonstrating that local detonations cannot occur in lieu of evaluating the effects of local detonations.

Comment 8: Licensees should not have to justify the selection of a safety system. Only the adequacy of the system should be of concern.

Comment 9: The particular analysis method to be used should be left optional so that the approach can be selected by the licensee to fit his particular capabilities and specific plant design.

Comment 10: The proposed hydrogen release rate of 1000 lb/min is not supported by current data for recoverable cores and should not be specified for the final rule.

29. Texas Utilities Generating Co. - Utility

Comment 1: The requirement for equipment to achieve cold shutdown should be deleted. Cold shutdown should be addressed in the same manner as it has in the past with the added consideration of the proper hydrogen conditions.

Comment 2: There is no technical justification for requiring a demonstration of either survivability or qualification of equipment for a postulated hydrogen burn. It is anticipated that the implementation of the rule would impose a severe burden on the industry with no evidence of a significant safety problem.

30. GPU Nuclear - Utility

Comment 1: In view of the improvements that have been made since the TMI-2 accident that reduce the likelihood of a DCA, it is inappropriate to implement a requirement that addresses the hydrogen burn issue for PWR large dry containments.

Comment 2: The issue of equipment qualification for hydrogen burn conditions should be kept separate from the qualification of equipment for current DBAs.

Comment 3: The two phased approach for equipment survivability can result in an unwarranted considerable financial impact. Only one set of criteria should be implemented.

Comment 4: Since both accident sequences and hydrogen generation rates will be plant specific, the first approach is most appropriate and realistic. However. the sequences should be prioritized using PRA and only the most likely ones analyzed.

Comment 5: The 75% clad reaction rate is unrealistic based on the results of TMI-2 and analysis of recovered degraded core accidents.

Comment 6: In view of the extremely small probability of occurrence of local detonations in nuclear plants, due to uniform mixing, relatively open geometries and containment sprays, it is unreasonable to require survivability of essential equipment after being exposed to local detonations.

Comment 7: The differences in the definitions of "survivability" and "qualification" standards are too vague. "Survivability" should include no margins or conservatisms. "Survivability" criteria are all that is needed for a low prabability event as a DCA.

Comment 8: The design basis for all plants is safe shutdown. Depending on individual licensee commitments, this can be either hot or cold shutdown. Unless the word "cold" was deleted it would mandate a new design basis for many plants.

33. Louisiana Power and Light - Utility

Comment 1: Since ECCS degradation is the governing event in significant hydrogen release scenarios, it would be more appropriate for the rule to codify the extent of ECCS degradation required to be postulated rather than the percentage of fuel clad oxidation. The rule should require the control of that amount of hydrogen resulting from degradation of the ECCS for a period of time to be based on the reliability of the ECCS. Sensitivity analyses would be required to determine the accident scenario producing the worst case hydrogen generation.

Comment 2: The 75% metal-water reaction is not credible since conservative analyses indicate that the core would have to be uncovered for 16 hours. It is not reasonable to expect the core to be uncovered for such a length of time.

Comment 3: The 75% metal-water reaction should not be considered to include contributions from radiolysis and other sources since radiolysis does not provide a major hydrogen contribution until 2-3 days after the accident.

Comment 4: It is not clear what the intent is regarding the two-step equipment qualification plan. How will the environmental conditions differ between Es and Eq? How will criteria be established for deciding which equipment meets Es or Eq?

Comment 5: The 75% metal-water reaction is not consistent with an accident such as occurred at TMI-2. Such a large metal-water reaction would have consequences for exceeding that of TMI-2.

Comment 6: The first suggested approach appears the most reasonable, however, PRA techniques should be permitted for determining the magnitude of hydrogen generation as well as the release rate.

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Hydrogen Control Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Final Rule.

SUMMARY: The Commission is amending its regulations to improve hydrogen control capability for boiling water reactors with MARK III containments and for pressurized water reactors with ice condenser containments. The amendments require improved hydrogen control systems that can handle large amounts of hydrogen during and following an accident. For those of the above reactors not relying upon an inerted atmosphere for hydrogen control, the rule requires that certain systems and components be able to function during and following hydrogen burning. The rule also requires affected licensees to submit analyses to the Commission in support of the previous two requirements. The rule is needed to improve the capability of some types of nuclear power reactors to withstand the effects of an accident like the one which occurred at Three Mile Island. The new requirements will result in greater assurance that nuclear power reactors can be safely shut down following a Three Mile Island type of accident.

EFFECTIVE DATE:

FOR FURTHER INFORMATION CONTACT: Morton R. Fleishman, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone 301-443-5997.

SUPPLEMENTARY INFORMATION:

Background

The Commission has taken numerous actions to correct the design and operational limitations that were revealed by the accident at Three Mile Island, Unit 2 (TMI-2), which resulted in a severely damaged or degraded reactor core, in a concomitant release of radioactive material to the primary coolant system, and in a fuel cladding-water reaction causing the generation of a large amount of hydrogen. Included in these actions are several rulemaking proceedings intended to improve the hydrogen control capability of light-water nuclear power reactors.

On December 23, 1981, the Commission published in the Federal Register (46 FR 62281) a notice of proposed rulemaking on "Interim Requirements Related to Hydrogen Control," inviting written comments or suggestions on the proposed rule by February 22, 1982. A notice extending the comment period for an extra 45 days to April 8, 1982 including editorial corrections was published in the Federal Register on February 25, 1982 (47 FR 8203). The notice concerned proposed amendments to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which would have required that:

a. Each boiling water reactor (BWR) with a Mark III type containment and each pressurized water reactor (PWR) with an ice condenser type containment be provided with a hydrogen control system capable of handling an amount of hydrogen equivalent to that which would be generated if there were at least a 75 percent fuel cladding-water reaction without loss of containment integrity;

b. Each boiling water reactor and each pressurized water reactor that does not rely on an inerted atmosphere for hydrogen control be provided with safety systems needed to establish and maintain safe cold shutdown and maintain containment integrity that can function after the burning of substantial amounts of hydrogen; and

c. Analyses be performed for the reactor categories mentioned above to justify the hydrogen control systems selected and to assure containment structural integrity and survivability of needed safety systems during a hydrogen burn.

It should be noted that the proposed rule was not part of the separate, long-term rulemaking on degraded or melted cores (the "severe accident rulemaking") for which an advance notice of proposed rulemaking was published on October 2, 1980 (45 FR 65474) and which was the subject of SECY 82-1B[1], "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation."

General Comments

Twenty-eight persons submitted comments regarding the proposed amendments. The comments and the SECY Paper noted above are part of the public record and may be examined and copied, for a fee, in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C. A summary of the comments amd a comment analysis are also available for inspection and copying, for a fee, in the Public Document Room.

The comments received have been carefully reviewed and evaluated during preparation of this final rule. The final rule contains revisions to the proposed rule that reflect consideration of these comments. The commenters generally provided many specific comments on all aspects of the proposed amendments. The following discussion represents a distillation of the more significant comments.

[1] SECY 82-1, dated January 4, 1982 and SECY 82-1B, dated November 24, 1982 are available for inspection at the Commission's Public Document Room at 1717 H Street, NW,

Numerous commenters suggested that the implementation of the Hydrogen Control Rule should be deferred until the severe accident rulemaking (see above) when applicable research and probabilistic risk analyses (PRAs) will be completed. The Commission agrees with these comments relative to PWRs with large dry containments. Dry containment designs have a greater inherent capability to accommodate large quantities of hydrogen because of their higher design pressure and larger volume; therefore, for these designs the Commission believes that rulemaking with regard to hydrogen control can be safely deferred pending completion of NRC- and industry-sponsored research. Furthermore, with regard to systems and components that must be able to function during and following hydrogen burning, the results of the TMI-2 containment survey indicate that such systems and components did function properly following the burn event.

With regard to BWRs with Mark III containments and PWRs with ice condenser containments, the Commission believes that these containments can safely accommodate the burning of hydrogen from about a 25% metal-water reaction.[2] However, since the TMI-2 accident showed that a 45-50% metal-water reaction was possible, the Commission believes that it is necessary to enhance the hydrogen control capability for reactors with these types of containments and that new regulations are required to ensure that the proper design features are incorporated. Adoption of the final rule will also formalize Commission regulatory decisions currently being applied on a case-by-case basis in individual licensing proceedings and will provide the needed basis for regulatory actions that cover licensing and continued operation of the affected plants. Additionally, this rule is intended to remove the questions of hydrogen control and the ability of certain systems and components to function after a hydrogen burn as items of litigation in individual proceedings.

2 The basis for this belief is contained in SECY 80-107, (from pg. 5) which is available for inspection at the Commission's Public Document Room at 1717 H Street, NW, Washington, D.C.

Several commenters stated that the 75 percent metal-water reaction required to be assumed for design and analysis is unreasonably high based on evaluation of the TMI-2 accident and analyses of recoverable degraded core accidents.[3] The 75 percent metal-water reaction chosen by the Commission is significantly greater than that which occurred during the TMI-2 accident; however, the primary intent of the rule is to require containment designs that can accommodate accident sequences in which hydrogen combustion poses the principal threat to containment integrity. Consequently, the Commission believes it is prudent to specify a value sufficiently greater than that which was analyzed to have occurred at TMI-2 so that there will be an appropriate margin of safety. In this regard, it should be noted that the 75% value refers only to the cladding surrounding the active fuel region. Not all of the zirconium which can interact is in this fuel cladding. For example, BWR channel box temperatures may be close to the cladding temperature, just as the grid spacers in PWRs and BWRs will be. All these items contain zirconium and the intent of the 75% value is to account for reactions in these items as well. The Commission feels confident that the 75 percent value is representative of a limiting case degraded core accident (beyond which a core melt is likely to occur). Finally, the Commission sees no significant benefit in reducing the metal-water reaction to a level such as 50 percent for those plants required to install a hydrogen control system since the basic design of the system would not change.

3 See the following studies ~~are~~ available for inspection at the Commission's Public Document Room at 1717 H Street, NW, Washington, D. C.:

NUREG/CR-2540, "A Method for the Analysis of Hydrogen and Steam Releases to Containment During Degraded Core Cooling Accidents," February 1982

NUREG/CR-1219, "Analysis of the Three Mile Island Accident and Alternative Sequences," January 1980

"Report on Hydrogen Control Accident Scenarios, Hydrogen Generation Rates and Equipment Requirements, "Rev. 1, July 1982 - Submitted by the BWR/6 MARK III Hydrogen Control Owners Group.

move to p. 4 3 SECY 80-107, "Proposed Interim Hydrogen Control Requirements for Small Containments," February 22, 1980.

A number of commenters recommended that the requirement for a hydrogen control system be revised to permit licensees the option of demonstrating analytically that additional hydrogen control systems are not necessary because of intrinsic design capabilities that reduce the likelihood of hydrogen generation. While it is true that design features to reduce hydrogen generation are necessary and desirable, the Commission still believes that, in order to cope with unexpected events, there should be a solution to the hydrogen issue that involves design features that ensure containment integrity, even if a large amount of hydrogen is generated. Thus, while measures to prevent the the generation of large amounts of hydrogen are necessary and desirable, the Commission believes that it is also necessary, depending upon containment design, to provide measures to mitigate the effects of large amounts of hydrogen.

Some commenters indicated, that, since the primary function of the containment is to prevent excessive radiation dose to the public, the rule should be modified to preclude the loss of containment function rather than to preclude the loss of containment integrity. The Commission appreciates the fact that some nuclear plants are designed with a multi-building, multi-barrier concept that is intended to prevent the leakage of radiation by diverse methods such as filtering and scrubbing mechanisms, plate-out mechanisms, and containment sprays. However, the Commission's basic and long-standing safety philosophy remains the same, namely, that the containment should be designed to remain intact following an accident in order to provide additional assurance that excessive radiation will not be released. In other words, the Commission reaffirms its policy that the prevention of excessive radiation dose to the public can best be assured by maintaining a leak tight containment and that this, in turn, can be provided by assuring that there is structural integrity with margin.

Some commenters stated that the criterion for containment structural integrity is unnecessarily restrictive. They stated that it should not be limited to the provisions of the ASME Boiler and Pressure Vessel Code, but should permit the use of other methods such as realistic analyses using actual material properties. The Commission agrees with this comment and has modified the rule in this regard. Section 50.44(c)(3)(iv) has been changed to indicate that "containment structural integrity must be demonstrated by use of a method that has been ~~previously~~ accepted by the NRC staff." The rule includes two alternative methods as examples but does not preclude other methods that may be shown to be acceptable to the Commission.

It was suggested by some commenters that the rule should address only non-inerted, small-volume, low-pressure containments and should not impose requirements on the remaining containments since for these latter ones it would provide, at best, insignificant improvements in safety. The Commission agrees for the reasons indicated above; therefore, as indicated previously, it has revised the rule to apply only to Mark III BWRs and ice condenser PWRs.

A number of commenters stated that the rule ignores those post-TMI suggested improvements which have been implemented and which reduce the likelihood of a degraded core accident. In the case of PWRs with large dry containments, as discussed above, the Commission ~~feels~~ believes that the post-TMI improvements, along with the inherent strength of the containments, as discussed above, have indeed provided sufficient safety to permit the delay of any additional rulemaking until completion of ongoing research programs.

It has been recommended that in view of the small probability of occurrence of local detonations as a result of various design features, the rule should permit licensees the option of demonstrating that local detonations cannot occur in lieu of evaluating the effects of local detonations. The Commission agrees and has modified paragraphs 50.44(c)(3)(v) and (vi) of the rule appropriately.

Many commenters indicated that they believe the requirement that systems and components that can function after a hydrogen burn be provided for "safe cold shutdown" is unnecessary and is inconsistent with the licensing basis for most operating plants which requires only "safe shutdown". Those commenters felt that the safe shutdown criterion should not be an issue with regard to hydrogen control, but that it should be considered in another forum. Because of the fact that a degraded core accident is less likely than a design basis accident, the Commission agrees that the requirement for cold shutdown may be overly conservative. The licensing basis for most plants is, in fact, just safe shutdown. The reference to cold shutdown has been deleted from the rule; but the Commission notes that the issue of safe shutdown versus safe cold shutdown has not yet been resolved. The issue is expected to be addressed within the context of the resolution of Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," which is the subject of current NRC staff effort.

Several commenters have suggested that the implementation schedules should be made more realistic so that design changes logically follow after the required analyses are completed. The Commission agrees. The greatest relief, of course, has come by deferring implementation of the rule for PWRs with large dry containments. However, the rule has also been revised to specify that the required analyses be submitted to the Commission within one year and the corresponding design changes be completed within two years.

Some commenters noted that in the Supplementary Information accompanying the proposed rule it was stated that the selection of the hydrogen control system should be supported by comparative analyses of alternative systems to show their relative advantages and disadvantages. They stated that this guidance is inconsistent with Commission practice and is unnecessary. They felt that the only requirement should be a demonstration that the selected system is suitable for its intended application.

The Commission agrees that this guidance was inconsistent with Commission practice in the case of operating reactors and those whose operating licenses are about to be issued in the near-term. In the final rule, § 50.44(c)(3)(vi) has been modified to delete the implication that comparative analyses are required and to indicate that the analysis is intended to support the design of the hydrogen control system that is selected. Comparative analyses of alternative systems are not required.

HYDROGEN CONTROL SYSTEMS [§ 50.44(c)(3)(iv]

As originally proposed, applicants and licensees with boiling water reactor (BWR) facilities with Mark III type containments and pressurized water reactor (PWR) facilities with ice condenser type containments, for which construction permits were issued prior to March 28, 1979, are required to install hydrogen control systems capable of accommodating an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding (surrounding the active fuel region) with water, without loss of containment integrity. The particular type of hydrogen control system to be selected is left to the discretion of the applicant or licensee; however, the NRC must find it acceptable based upon suitable programs of experiment and analysis. The design of the selected system must be supported by the analyses which are to be submittted as part of the analyses required under § 50.44(c)(3)(vi). The system that is proposed and approved must safely accommodate large amounts of hydrogen, and operation of the system, either intentionally or inadvertently, must not further aggravate the course of an accident or endanger the plant during normal operations. As discussed previously, the amount of hydrogen to be assumed in the design of the hydrogen control system is that amount generated when 75% of the fuel cladding surrounding the active fuel region reacts with water.

As discussed above, the limited method proposed to demonstrate containment structural integrity has been expanded. Containment structural integrity may now be demonstrated by use of a method previously accepted by the NRC staff. One of the acceptable methods is the use of the applicable ASME Boiler and Pressure Vessel Code. However, the Commission will accept other methods, provided that convincing evidence is presented regarding their suitability.

Other changes from the proposed rule are the relaxation of the implementation date to two years rather than one year after the effective date of the rule and the elimination of the word "cold" in the phrase "safe cold shutdown."

SYSTEMS AND COMPONENTS ~~CONSIDERED UNDER~~ [§ 50.44(c)(3)(v)]

At the time the proposed rule was issued for comment, the Commission indicated that it was considering a two-step approach to address "qualification" (as defined below) of those systems and components that must be able to function during and after a hydrogen burn. For the reasons explained below, the Commission did not choose this two-step approach. As the proposed first step, there would have been a demonstration that these systems and components could "survive" the hydrogen burn and continue to be able to perform their safety function. This step would not have entailed that these systems and components actually be qualified pursuant to NRC's qualification program. The proposed second step, would have entailed the actual "qualification" of these systems and components. The conceptual differences between systems and components demonstrated to be "survivable" and systems and components demonstrated to be "qualified" were also described.

The Commission specifically sought comments on the use of the two-step approach for defining standards, on the "survivability" and "qualification" approaches themselves, and on proposals for implementation schedules. There were numerous comments in response to this request. The overwhelming reaction was that the two-step approach to reaching a survivability determination is unwarranted and will unnecessarily escalate the costs to industry. Many commenters felt that a straightforward survivability approach would be appropriate provided reasonable guidelines are specified. In view of the smaller likelihood of a degraded core accident as compared to a design basis accident, which has been reduced further by post-TMI improvements, the Commission has decided to forego the two-step approach previously described. The Commission now believes, in view of the recent issuance of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety," that there is no significant difference between demonstrating survivability and demonstrating qualification. Paragraph (f) of § 50.49 describes several methods, one of which must be used, for qualifying electrical equipment important to safety. For example, for those licensees which have already demonstrated survivability, as described in the Supplementary Information of the notice of proposed rulemaking for this rule on hydrogen control requirements (46 FR 62281, Dec. 23, 1981), the qualification methods described in paragraphs (f)(2) and (f)(4) of § 50.49 could be used to show that the systems and components have been qualified. In this regard, the margins considered adequate for a degraded core accident are less than those considered adequate for a design basis accident due to the lower probability of occurrence of a degraded core accident.

The Commission now views "qualification" as the generation and maintenance of evidence using tests and analyses to assure that systems and components will operate on demand to meet system performance requirements. In the case of a hydrogen burn environment, this means that there must be evidence that systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity are capable of performing their functions during and after exposure to the environmental conditions created by the postulated accident including the burning of hydrogen. Qualification may be demonstrated in a manner acceptable to the Commission using a combined approach of analysis and testing. Thus, an acceptable thermal analysis would have to be performed for the containment in order to determine the thermal response of the systems and components during a hydrogen burn. This thermal response should then be compared to the thermal response the systems and components had during their qualification testing. The licensee should then demonstrate that the qualification thermal response envelops the thermal response during a hydrogen burn. Selected tests should also be performed at predicted hydrogen burn conditions (or, other tests previously performed may be referenced if demonstrated to be applicable) to convince the Commission that the systems and components are qualified to perform their functions during and following a hydrogen burn.

Paragraph 50.44(c)(3)(v) applies to those Mark III BWRs and ice condenser PWRs that do not have an inerted containment atmosphere for hydrogen control. At present, this includes all Mark III BWRs and ice condenser PWRs, since no applicant or licensee has as yet elected to use the inerting option for these plants. The systems and components that must be qualified for a hydrogen burn are those needed (a) to shut down the reactor and bring it to and maintain it in a safe shutdown condition, and (b) to prevent loss of containment integrity. These systems and components can be further categorized as follows:

a. Systems and components mitigating the consequences of the accident;

b. Systems and components needed for maintaining integrity of the containment pressure boundary;

c. Systems and components needed for maintaining the core in a safe condition; and

d. Systems and components needed for monitoring the course of the accident.

As discussed previously, these systems and components are described as bringing the reactor to "safe shutdown" rather than" safe cold shutdown." Furthermore, the time for implementation has been changed to two years rather than one year. Finally, the rule has been revised to indicate that the environmental conditions to be assumed for a hydrogen burn do not have to include the effect of local detonations if it is shown to the Commission's satisfaction that local detonations are unlikely to occur.

ANALYSES [§ 50.44(c)(3)(vi)]

In the proposed rule, the Commission included a description of three different approaches concerning the supplementary guidance to be provided for performing the required analyses for the design of the hydrogen control system. These were (a) analyses of different accident scenarios, (b) analyses of a single accident scenario with variation of key parameters, and (c) analyses using an "envelope of time histories of hydrogen and steam release rates" to be supplied by the Commission. The Commission requested comments concerning which of the approaches was preferred as well as suggestions regarding improvements or other alternatives.

There was no preponderance of comments leaning toward a particular approach; however, the first two approaches appeared to have greater support. Furthermore, many commenters felt that there should be flexibility in the approach to be used and in the selection of the accident scenarios. It was also suggested that the accident scenarios should be considered in order of importance using PRA techniques.

Based on the comments received, the Commission has decided that it will not choose between the first two approaches, and that licensees need not use the third approach. It is left to each licensee to suggest to the Commission which of the first two approaches it wishes to use and to arrive at a mutually agreeable method with the Commission for performing the analyses.

Either of the following two approaches may be used for performing the analyses. However, licensees are <u>not restricted</u> to the specified scenarios. If, because of unique plant design features, other scenarios are known to present a greater likelihood of core degradation than those identified by the Commission, the analyses should be based on the scenarios known to present the greatest likelihood of core degradation. For example, if for a particular plant an intermediate break loss of coolant accident (LOCA) results in a greater likelihood for core degradation than the scenarios in Table I, the licensee should base its calculations on the intermediate break LOCA scenario.

In the first approach, the Commission has selected accident sequences or scenarios which have been found by PRA techniques to be significant contributors to the likelihood of core degradation and thus pose a significant hydrogen threat. If it selects this approach, the licensee should perform analyses, using these sequences, to determine the time variation of the hydrogen and steam release rates to the containment building. The analyses (which should include the failure-assumptions of the different scenarios, as well as the accident recovery phase and allowances for uncertainties) should provide the pressure and temperature histories to which the containment will be exposed. A suggested list of accident sequences to be used in this approach is given in Table I. The scenarios include the production of substantial amounts of hydrogen as part of core-melt sequences, these scenarios were selected, based on experience and engineering judgment, because they are representative of the more probable severe accident sequences which could be terminated short of primary vessel melt-through with available recovery techniques.

In the second approach, a base sequence will be identified by both the Commission and licensee based on its significance and characteristics from the standpoint of hydrogen threat. Key aspects of this scenario should then be parametrically varied by the licensee in determining the acceptability of the hydrogen control system or the containment response. This will provide a wider range of parameters than that of the selected base sequence alone. The acceptablility of the analyses used in this approach depends on the selection and range of the parameters being varied. The range must be chosen to include the effects of recovering from the degraded condition. If a licensee has determined that, because of its own plant design, another scenario presents

Table I. Accident Sequences Leading to a Significant Hydrogen Threat

PWR

1. Small LOCA with temporary loss of emergency core cooling (ECC) injection.

2. Transient with temporary loss of all feedwater and the high pressure ECC system.

3. Interruption of all AC electric power with failure of the auxiliary feedwater system.

BWR

4. Transient with reactor isolation and temporary failure of all coolant make-up systems.

5. Small LOCA with temporary failure of ECC injection.

6. Transient with failure of reactor shutdown systems and interruption of ECC systems.

Table II. Parametric Variations of a PWR Small-Break Scenario

Rate of H_2 Release[1] [lb/min]	Timing of H_2 Release	Rate of Steam (Enthalpy) Release [lb/min (millions of Btu/min)][2]	Concurrent Failures and Recoveries[3]
2	- Starting at the time the top of the core is uncovered	- 600(1)	- Fans
10	- Before major steam release	- 3,600(6)	- Containment sprays
30	- Concurrent with major steam release	- 10,000(16)[4]	- All AC power
100	- After major steam release		- Recirculation
150[5]			

[1]These rates should be assumed to be constant during the period of release and represent release from the primary system to the containment building.

[2]The conversion from mass rate to enthalpy rate is based on 1600 Btu/lb which is believed to be appropriate for steam superheated by excessively hot fuel.

[3]These items are intended to be applied, as appropriate, in either the faster or the more sustained hydrogen releases, and are not necessarily to be applied for each variation considered.

[4]This high rate of steam release may occur for about 10 min. during ECC recovery.

[5]See NUREG/CR-2540, previously referenced, for a discussion.

a greater likelihood of core degradation than the small break LOCA, that scenario should be chosen for parametric study. The variables and values studied should be determined case-by-case depending on the particular scenario. Table II represents a list of parameter variations that provide reasonable extensions of a PWR small-break scenario (Item 1 of Table I). It should be noted that the maximum hydrogen release rate in the first column has been reduced to 150 lb/min., from the 1000 lb/min. value originally proposed, as a result of new analyses and several comments.

REGULATORY ANALYSIS

The Commission has prepared a regulatory analysis for this regulation. The analysis examines the costs and benefits of the rule as considered by the Commission. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, NW, Washington, DC. Single copies of the analysis may be obtained from Morton R. Fleishman, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone (301) 443-5997.

PAPERWORK REDUCTION ACT

This final rule imposes information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501, *et seq.*) These requirements were approved by the Office of Management and Budget. Approval Number 3150-0011.

Enclosure "F"

REGULATORY FLEXIBILITY ACT

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this rule does not fall within the purview of the act.

LIST OF SUBJECTS IN 10 CFR PART 50

Antitrust, Classified information, Fire prevention, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, and Reporting requirements.

Accordingly, notice is hereby given that, pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of Title 5 of the United States Code, the following amendments to 10 CFR Part 50 are published as a document subject to codification.

PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183. 186, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.58, 50.91 and 50.92 also issued under Pub. L. 97-415, 96-Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Section 50.100-50.102 also issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.10(a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10(b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)) and §§ 50.55(e), 50.59(b), 50.70, 50.71, 50.72, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. In § 50.44, paragraph (c)(3) is revised by adding new paragraphs (iv), (v) and (vi) to read as follows:

§ 50.44 Standards for combustible gas control system in light water cooled power reactors.

* * * * *

(c)(3) ***

(iv)(A) Effective [insert a date two ~~[one]~~ years after the effective date of the amendment ~~rule~~], or the date of issuance of a license authorizing operation above 5% [~~percent~~] of full power, whichever is later, each licensee which has a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee which has a pressurized light-water nuclear power reactor with an ice condenser type of containment, ~~[for which]~~ issued a construction permit [~~was issued prior to~~] before March 28, 1979, shall ~~[be]~~ provide~~[s]~~ its nuclear power reactor with a[~~n acceptable~~] hydrogen control system justified by a suitable program~~[s]~~ of experiment and analysis. The hydrogen control system must be capable of handling without loss of containment structural integrity an amount of hydrogen equivalent to that generated from a metal-water ~~[the]~~ reaction ~~[of]~~ involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). ~~[with water]~~

(B) Containment structural integrity must be demonstrated by use of a method ~~previously~~ [illegible] accepted by the NRC staff. This method could include the use of actual material properties with suitable margins to account for uncertainties in modeling, in material properties, in construction tolerances, and so on. Another method could include a showing that the following specific criteria of the ASME Boiler and Pressure Vessel Code are met:

(1) That steel containments ~~[must]~~ meet the requirements of the ASME Boiler and Pressure Vessel Code (Edition and Addenda as incorporated by reference in paragraph 50.55a(b)(1) of this part), specifically in Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, [~~except that~~] considering pressure and dead load alone (evaluation of instability is not required); and

(2) That concrete containments ~~[must]~~ meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone.

(C) Subsubarticle NE-3220, Division 1, and subsubarticle CC-3720, Division 2, of Section III of the ASME Boiler and Pressure Vessel Code, referenced in paragraphs (c)(3)(iv)(B)(1) and (c)(3)(iv)(B)(2) of this section, [~~These subsubarticles~~] have been approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H Street NW., Washington, DC.

(D) If the hydrogen control system relies on post-accident inerting, the containment structure must be capable of withstanding the increased pressure:

(1) During the accident, where it [~~must~~] is acceptable to show that it does not exceed Service Level C Limits or the Factored Load Category (as [~~previously specified~~] described in paragraph (c)(3)(iv)(B) of this section [~~paragraph~~]); and

(2) Following inadvertent full inerting [~~that may occur~~] during normal plant operations, where it [~~must~~] is acceptable to show that it does not exceed either the Service Level A Limits of Subsubarticle NE-3220 (for a steel containment) or the Service Load Category of Subsubarticle CC-3720 (for a concrete containment).

Insert A from next page

(3) ~~(E)~~ If the hydrogen control system relies on post-accident inerting, the systems and components [~~equipment~~] required to establish and maintain safe [~~cold~~] shutdown and containment integrity must be designed and qualified for the environment caused by such [~~post-accident~~] inerting. Furthermore, inadvertent full

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inerting during normal plant operations must not adversely affect systems and components needed for safe operation of the plant. Modest deviations from these criteria in this paragraph will be considered by the Commission if good cause is shown. A

(v) (A) Effective [insert a date two years after the effective date of the amendment], or the date of issuance of a license authorizing operation above 5% of full power, whichever is later, each licensee which has a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee which has a pressurized light-water nuclear power reactor with an ice condenser type of containment, [~~for-which~~] issued a construction permit [~~was-issued-prior-to~~] before March 28, 1979 for a reactor that does not rely upon an inerted atmosphere to control hydrogen inside the containment, shall [~~be~~] provide[~~d~~] its nuclear power reactor with systems and components necessary to establish and maintain safe [~~cold~~] shutdown and to maintain containment integrity. These systems and components must be [~~that-are~~] capable of performing their functions during and after [~~being-exposed~~] exposure to the environmental conditions created by the burning [~~(or-local-detonation)~~] of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur.

(B) The amount of hydrogen to be considered is equivalent to that generated from [the] a metal-water reaction [~~of~~] involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). [~~with-water---This-requirements-shall-be-effective-as-follows: for-each-boiling-light-water-nuclear-power-reactor-with-a-Mark-III-type-containment and-each-pressurized-light-water-nuclear-power-reactor-with-an-ice-condenser-type-containment,-on-[one-year-after-the-effective-date-of-the-rule]-or-the date-of-issuance-of-a-license-authorizing-operation-above-5-percent-of-full power;-whichever-is-later;-for-every-other-light-water-nuclear-power-reactor~~]

~~that-must-meet-this-requirement;-on [two years after the effective date of the rule]-or-the-date-of-issuance-of-a-license-authorizing-operation-above 5-percent-of-full-power;-whichever-is-later.~~]

(vi) (A) Effective [insert a date one year after the effective date of the amendment], or the date of issuance of a license authorizing operation above 5% of full power, whichever is later, each licensee which has a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee which has a pressurized light-water nuclear power reactor with an ice condenser type of containment, issued a construction permit before March 28, 1979, shall [~~analyses-shall-be-performed-and~~] submit[~~ted~~] an analysis to the Director of the Office of Nuclear Reactor Regulation. [~~for-each-light-water-nuclear-power-reactor;-for-which-a-construction-permit-was-issued-prior-to-March-28;-1979;-to-evaluate~~]

(B) The analysis required by paragraph (c)(3)(vi)(A) of this section must:

(1) Provide an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75% [~~percent~~] of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume) and include [~~with-water-including~~] consideration of hydrogen control measures as appropriate; [~~Each-analysis-must~~]

(2) Include the period of recovery from the degraded condition;

(3) Use [~~the~~] accident scenarios [~~to-be-used-in-the-analyses-must-be acceptable-to-the-NRC-staff;~~]-[~~The-scope-and-implementation-requirements-for-the-analyses-for-the-various-types-of-light-water-nuclear-power-reactors-are as-follows:~~]

~~(A)--For-each-boiling-light-water-nuclear-power-reactor-with-a-Mark-III-type containment-and-each-pressurized-light-water-nuclear-power-reactor-with-an--~~

~~ice condenser type containment, analyses shall be performed that justify the selection~~]

(4) Support the design of the hydrogen control system selected [~~required by § 50.44~~] under paragraph(c)(3)(iv) of this section; and, [~~These analyses shall be completed and submitted by [one year after the effective date of the rule or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later.~~]

(5) Show that, for those reactors described in paragraph (c)(3)(iv) of this section that do [~~For each light-water nuclear power reactor that does~~] not rely upon an inerted atmosphere to control hydrogen inside the containment: [~~analyses shall be performed to show that~~]

(i) The containment structural integrity, as [~~defined~~] described in [~~§ 50.44~~] paragraph (c)(3)(iv) of this section, will be maintained; and

(ii) Systems and components necessary to establish and maintain safe [~~cold~~] shutdown and to maintain containment integrity will be capable of performing their functions during and after [~~being exposed~~] exposure to the environmental conditions created by the burning of hydrogen, including the effect of local detonations, unless such detonations can be shown unlikely to occur. [~~These analyses shall be completed and submitted as follows: for each boiling light-water nuclear power reactor with a Mark III type containment and each pressurized light-water nuclear power reactor with an ice condenser type containment, by [one year after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later; for every other light-water nuclear power reactor for which these analyses are required, by [two years after the effective date of the rule]~~

~~or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later.]~~

Dated at Washington, D.C. this _______ day of _________, 1983.

For the Nuclear Regulatory Commission,

_______________________________.
Samuel J. Chilk
Secretary of the Commission

Enclosure "F"

REGULATORY ANALYSIS
FOR
AMENDMENTS RELATED TO HYDROGEN CONTROL

1. STATEMENT OF THE PROBLEM

1.1 Background

The accident at Three Mile Island, Unit 2 (TMI-2) resulted in a severely damaged or degraded reactor core, a concomitant release of radioactive material to the primary coolant system, and a fuel cladding-water reaction which resulted in the generation of a large amount of hydrogen. The Commission has taken numerous actions to correct the design and operational limitations revealed by the accident. Included in these actions are several rulemaking proceedings intended to improve the hydrogen control capability of light-water nuclear power reactors. On October 2, 1980, the Commission published in the Federal Register (45 FR 65466) a notice of proposed rulemaking on "Interim Requirements Related to Hydrogen Control and Certain Degraded Core Considerations" (Interim Rule). The notice concerned proposed amendments to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to improve hydrogen management in light-water reactor facilities and to provide specific design and other requirements to mitigate the consequences of accidents resulting in a degraded reactor core.

On March 23, 1981, the Commission published in the Federal Register (46 FR 18045) a notice of proposed rulemaking on "Licensing

Requirements for Pending Construction Permit and Manufacturing License Applications." The notice proposed a set of licensing requirements applicable to construction permit applications that stemmed from lessons learned from the TMI-2 accident. On May 13, 1981, the Commission published in the Federal Register (46 FR 26491) a notice of proposed rulemaking on "Licensing Requirements for Pending Operating License Applications" (OL Rule).

As a follow-up to the October 2, 1980 notice of proposed rulemaking, the Commission published a notice of final rulemaking on December 2, 1981 (46 FR 58484) on hydrogen control requirements related to inerting of Mark I and II boiling water reactors, hydrogen recombiner capability and high point vents.

The Commission has also been considering the ability of all light-water nuclear power reactors, particularly pressurized light-water reactor facilities with ice condenser-type containments and boiling light-water reactor facilities with Mark III-type containments, to withstand an accident with the concomitant generation of large amounts of hydrogen, such as the type which occurred at Three Mile Island, Unit 2 (TMI-2). As a result, three new amendments to the regulations were proposed for public comment via a notice of proposed rulemaking on December 23, 1981 (46 FR 62281). The amendments would require: (a) improved hydrogen control systems for boiling water reactors with Mark III type containments and for pressurized water reactors with ice condenser type containments; (b) that all light-water nuclear power reactors not relying upon an inerted atmosphere for hydrogen control show that certain important safety systems must be able to function during and following hydrogen burning; and finally (c) analyses to be submitted to justify the hydrogen control systems selected and to provide assurance that containment structural integrity will be maintained and important safety systems will continue to function following a hydrogen burn.

The Commission has required hydrogen control measures for ice condenser PWRs and for Mark III BWRs (for those that are operating and those that have pending operating license applications). The licensing actions taken are in basic agreement with the proposed amendments.

1.2 Description of Rulemaking

Section 50.44 of 10 CFR Part 50 is being amended to improve hydrogen control capability during and following an accident for BWRs with Mark III type containments and PWRs with ice condenser type containments. The amendments apply to those of the above reactors whose construction permit was issued prior to March 28, 1979 and would require:

a. hydrogen control systems that can handle large amounts of hydrogen,

b. certain systems and components and containment that are able to perform their functions during and following hydrogen burning, and

c. analyses to be performed and submitted that supports the design of the hydrogen control system selected and the demonstration of system and component survivability/ qualification.

As noted, the rulemaking requires submittal of the analyses to the Commission. The information contained in the analyses is necessary to permit the NRC staff to perform an evaluation to determine if the requirements for hydrogen control and system and component functioning during a hydrogen burn are met. Without this information the NRC staff could not evaluate the design of the hydrogen control systems or determine whether or not needed safety equipment could indeed function during a hydrogen burn.

2. OBJECTIVES

The objective of the rulemaking action is to provide specific requirements which, when implemented, will improve the capability of Mark III BWRs and ice condenser PWRs to withstand the consequences of a degraded core accident that generates a large amount of hydrogen.

The action will also formalize regulatory positions that have already been taken by the Commission in individual licensing cases (i.e., Sequoyah, McGuire, D.C. Cook and Grand Gulf).

3. ALTERNATIVES

The specific amendments are consistent with recent Commission licensing decisions. These decisions have been based on engineering evaluation and qualitative professional judgment that have evolved during the regulatory process. The technical decisions have been reviewed by the Advisory Committee on Reactor Safeguards (ACRS).

The rule is an outgrowth of recommendations made by the Lessons Learned Task Force (LLTF) in NUREG-0578, "TMI-2 LLTF Status Report and Short-Term Recommendations," and by the Commission in NUREG-0660, "TMI-2 Action Plan." It was recommended that short-term actions be implemented, in the form of rulemaking, to improve the capability of reactors to mitigate the consequences of degraded core accidents.

An alternative to rulemaking could be maintenance of the status quo with licensing decisions being treated on a case-by-case basis. However, this alternative would not result in any savings to NRC or industry since the requirements of the rule would still be implemented. In fact it would result in additional costs since it would leave the question of hydrogen control as an unresolved issue that would be subject to time-consuming and costly litigation for each case. For example, it has been estimated that the manpower cost in litigating the hydrogen control issue in the Perry hearing involves at least one man-year (my) from NRC and one my from the licensee, not including ASLB and intervenor costs. Hence, it was decided that the rulemaking route was the most reasonable alternative.

The rule does not require that any particular type of hydrogen control system be selected. However, as indicated in Section 4.2, the distributed igniter system is expected to be the system chosen to meet the requirements of the rule. Numerous other technical alternatives were considered during the development of the rule. These are:

a. Double-walled containments
b. Water fog sprays
c. Halon suppressants
d. Post-accident inerting
e. Inerting
f. Large capacity hydrogen recombiners
g. Purge systems
h. Filtered-vent systems.

While some of these systems are still under consideration, the distributed igniter system has advantages from the cost, operations, and reliability standpoint. For example, in the case of inerting, it is estimated that the initial capital costs alone for the 26 plants covered by the rule would be about $52,000,000 and the maintenance costs over the lifetime of the plants would be approximately $250,000,000. Furthermore, because of the frequency of containment entry for maintenance functions, especially for the ice chests in ice condenser types of containments, inerting is impractical. Air changes associated with each inert/de-inert cycle could cause excessive ice loss. Equipment reliability could also be affected by the restricted access as a result of the inerting. While BWRs with Mark III types of containments do not have the problem with the ice chests, much more equipment is located inside containment than for Mark I and II containments and thus the equipment reliability would be reduced by inerting.

Double-walled containments have been mentioned but not seriously considered because of the extremely high costs. It would involve essentially the construction of an additional large containment to surround the smaller containment so as to provide an increased volume to contain the generated hydrogen. In effect, adding a large dry containment. The cost of such an addition would be on the order of $400 million dollars per plant or $10 billion dollars for the 26 plants affected by the rule.

The Commission also considered pressurized water reactors with large dry containments for inclusion under the rule. However, because of the greater inherent capability of these plants to withstand the effects of hydrogen build-up as a result of their higher pressure capacity, larger volume, and the post-TMI improvements, it was decided to defer action on them until the completion of the long-term rulemaking on severe accidents.

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4. CONSEQUENCES

4.1 Plants Affected

There are currently 16 Mark III BWRs and 5 ice condenser PWRs in various stages of the licensing process and 5 operating ice condenser PWRs that are covered by the rule (i.e., whose construction permit was issued prior to March 28, 1979).

4.2 Costs of Hydrogen Control System

The cost of the hydrogen control system will clearly depend on the type of system selected, be it a distributed igniter system, a post-accident inerting system, or some other system. However, for the purpose of this analysis, it will be assumed that a distributed igniter system is selected since this is apparently the system of choice of licensees for both Mark III BWRs and ice condenser PWRs.

The cost of the equipment has been variously estimated as \$25,000 to \$140,000, with the lowest estimate supplied by a licensee who actually installed the equipment. The cost of installing the equipment. including QA costs, has been estimated by some to be \$50,000 and by others to be 5 my (\$500,000). There was one combined estimate of \$500,000 for equipment plus installation. The estimates for the design and analysis of the hydrogen control system have varied from 1.5 my (\$150,000) to \$750,000 (this included the survivability/qualification analyses and design work associated with the testing). For the purpose of this analysis, the equipment cost per plant will be assumed to be \$100,000, the installation cost will be \$300,000, and the design and analysis costs will be \$300,000. The installation costs are estimated for the forward fit of plants and would approximately double for backfits. However, the backfit would only be applicable to the five operating plants and they have already had the required modifications made.

Some of the Mark III plants have already begun installation of hydrogen control systems. For the purposes of the cost estimate, it will be assumed that 20 plants would be required to implement the rule giving costs as follows:

equipment = \$2,000,000 (20 x \$100,000)

installation = \$6,000,000 (20 x \$300,000)

design and analysis = \$6,000,000 (20 x \$300,000)

Total = \$14,000,000

It should be noted that some of the applicants have already taken steps for implementation based on interaction with the NRC staff.

4.3 Costs for Demonstration of Survivability/Qualification

The cost of implementing the survivability/qualification requirement will involve both analysis costs and costs of system and component testing. Much of the analysis performed in support of the hydrogen control system design is applicable to the demonstration of equipment survivability/qualification as well. It is estimated that the additional analyses required to meet the survivability/qualification requirement is 0.5 my (~ \$50,000) per plant.

The major systems and components tests needed will have been accomplished during the qualification program designed to qualify equipment for a LOCA environment. Some additional testing will be required for a hydrogen burn environment for certain items such as thermocouples and cables. The additional testing required for survivability/qualification is judged to be about $200,000 per plant.

The costs for the 20 plants are then:

testing = $4,000,000 (20 x $200,000)

analysis = $1,000,000 (20 x $50,000)

Total = $5,000,000

4.4 Costs of Analyses

The specific costs of this requirement are the analysis costs discussed under 4.2 and 4.3 and repeated here:

hydrogen control system	=	$6,000,000
survivability/ qualification	=	$1,000,000
Total	=	$7,000,000.

The cost of the reporting requirement for documenting and submitting the analyses to the Commission is included in the above figures and is estimated to represent about 10 percent of the total cost or $700,000.

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4.5 Sources of Cost Estimates

The above costs have been estimated based on preliminary data supplied by Pacific Northwest Laboratories (PNL) in support of Unresolved Safety Issue A-48, "Hydrogen Control Measures and Effect of Hydrogen Burns on Safety Equipment." Comments supplied by the Industry Degraded Core Rulemaking (IDCOR) Program and by Wisconsin Electric were also considered in arriving at the estimates. Finally, actual costs were solicited with regard to the Sequoyah, the McGuire and the Perry plants, since they have already had significant expenditures related to the design and installation of a distributed igniter system and the demonstration of survivability/qualification. These costs were tempered when arriving at the final estimated costs for the rule by the belief that the Sequoyah costs are expected to be higher than for future forward fitting of plants.

4.6 NRC Costs

The additional cost to the NRC is expected to result from the required evaluation of the submitted reports. It is estimated that it will involve about 24 man-weeks (mw) for the evaluation of each of the 20 reports for a total of 480 mw (9.2 my) or $920,000.

5. DECISION RATIONALE

Following the accident at TMI, the staff undertook a reassessment of plant capabilities to tolerate the consequences of a severe accident. One concern was that a hydrogen burn resulting from an accident similar to TMI-2 could result in a breach of containment. The staff concluded that ice condenser and Mark III type containments could safely accommodate the burning of hydrogen produced from a 25 percent fuel cladding-water reaction. However, since the accident at TMI-2 resulted in an estimated 45-50% reaction, it was felt prudent to require enhanced hydrogen control capability for reactors with these types of containments.

In 1981 the Commission began implementing the requirements, now being incorporated into this rule, for ice condenser PWRs and for Mark III BWRs. These requirements were intended to provide reasonable assurance, pending generic resolution of severe accident issues, that the risk of degraded core accidents for these types of plants is acceptable. Thus far the Commission has imposed these requirements on 5 plants in individual licensing cases following detailed plant reviews. The purpose of this rule is to codify the requirements already being imposed on plants on a case-by-case basis.

Although the requirement in this rule will result in substantial costs to the industry (~ $19,000,000) and the NRC (~ $900,000), the Commission has already determined in individual licensing cases that these requirements are necessary to assure acceptable levels of risk. The net result of codifying these requirements into the NRC regulations will be to eliminate the need for costly litigation of the hydrogen control issue in future licensing cases.

6. IMPLEMENTATION

6.1 Schedule

No implementation problems are now anticipated. As a result of comments received on the proposed rule, the schedule has been relaxed by at least one year. Now, only the analyses are required within one year; the corresponding design changes and survivability/qualification demonstration would not be required until two years.

6.2 Relationship to Other Schedules

In view of the implementation schedule recommended, it is not anticipated that other required actions will be affected since needed personnel can be acquired or reassigned to perform the tasks.

NRC ADOPTS ADDITIONAL HYDROGEN CONTROL REQUIREMENTS FOR NUCLEAR POWER PLANTS

The Nuclear Regulatory Commission is amending its regulations to improve the hydrogen control capability in nuclear power plants which have Mark III or ice condenser-type containments.

In the event of a loss-of-coolant accident, the cladding of the nuclear fuel could be damaged or melted and react with the reactor cooling water to form hydrogen. If sufficient quantities of hydrogen were released to a reactor containment and combine with oxygen, an explosion or fire could result in the loss of containment integrity and the subsequent release of large quantities of radioactivity to the environment.

The new amendments to Part 50 of the Commission's regulations require that owners of boiling water reactors with Mark III containments or pressurized water reactors with ice condenser containments assure that:

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--each reactor is provided with a system capable of handling--without loss of containment integrity--an amount of hydrogen equivalent to that which would be generated if at least 75 percent of the fuel cladding melted and reacted with cooling water;

--each reactor that does not rely on an inerted atmosphere (the oxygen in the atmosphere is replaced by a gas such as nitrogen) for hydrogen control, have safety systems--those systems necessary to establish and maintain a safe shutdown condition and maintain containment integrity--that can function after the burning of substantial amounts of hydrogen;

--analyses be performed for each reactor to support the design of the hydrogen control system and to assure the structural integrity of the containment and the survivability of needed safety systems during a hydrogen burn.

The new amendments are among a number of actions taken by the Commission since the March 1979 accident at Three Mile Island. That accident resulted in the generation of hydrogen--from the fuel cladding-water reaction--well in excess of the amounts assumed when the reactor containment was designed.

As a result of the accident, the NRC has initiated a long-term effort to determine to what extent nuclear power plants should be designed to deal effectively with accidents which result in damage to or melting of the nuclear fuel.

In the interim, however, the Commission determined that certain hydrogen control changes are of such safety significance that they should be implemented pending completion of the long-term effort. The initial measure requiring, among other things, inerted containments for boiling water reactors having Mark I and Mark II containments, was published in the Federal Register in December 1981.

The new amendments governing Mark III and ice condenser-type containments will become effective 30 days after publication in the Federal Register on _______________.

Implementation of the safety systems is required two years after the effective date of the amendments, or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later. The analyses requirement must be completed one year after the effective date of the rule, or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later.

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