T-1123

NUCLEAR REGULATORY COMMISSION

ORIGINAL

In the Matter of: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON EXTREME EXTERNAL PHENOMENA

DATE: August 11, 1982 PAGES: 1 - 55

AT: Washington, D. C.

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delete B. White

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400 Virginia Ave., S.W. Washington, D. C. 20024

Telephone: (202) 554-2345

UNITED STATES NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON EXTREME EXTERNAL PHENOMENA

Room 1406
1717 H Street, N.W.
Washington, D.C.

Wednesday, August 11, 1982

The Subcommittee met, pursuant to notice, at 1:00 p.m.

PRESENT:

DAVID OKRENT, Chairman
MYER BENDER
J. CARSON MARK
CHESTER P. SIESS
JESSE C. EBERSOLE
PAUL G. SHEWMON

NRC STAFF MEMBERS:

JIM KNIGHT
L. REITER
D. GUZY
D. NYOGI
A. BUSLIK
L. BERATAN

DESIGNATED FEDERAL EMPLOYEE:

R. SAVIO

P R O C E E D I N G S

MR. OKRENT: The meeting will now come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards, Subcommittee on Extreme External Phenomena.

I am David Okrent, the Subcommittee Chairman. Other ACRS members who are present or may be present during this meeting are Mr. Mark, Mr. Siess, Mr. Ebersole, perhaps Mr. Bender.

The purpose of the meeting is to discuss the ACRS recent recommendations on evaluation of seismic design margins for earthquakes more severe than the SSE.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act and the Government in the Sunshine Act. Dr. Richard Savio is the Designated Federal Employee for the meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register on Wednesday, July 21, 1982.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register notice. It is requested that each speaker first identify himself or herself and speak with sufficient clarity and volume so that he or she can be readily heard.

We have received no written statements from members of the public. We have no requests for time to make statements from members of the public.

It would seem to me, in looking over the material for this Subcommittee meeting, that the brief comment made by the ACRS in its report on Perry, in the reported dated July 13, 1982, perhaps is a way of opening up the subject to see what the staff has to say. In that report we said we recommend that the Applicant and the NRC Staff conduct studies to evaluate the margins to accomplish safe shutdown, including long-term heat removal following an earthquake of somewhat greater severity and lower likelihood than the safe shutdown earthquake. We believe it is important that there should be considerable assurance that the combination of seismic design basis and margins in the seismic design is such that this accident source represents acceptably low contribution to the overall risk from this plant.

It is the same thing as stated in similar letters, but I think that is maybe one succinct way of doing that.

Okay. Why don't we begin with Mr. Knight and see what the Staff has to say.

MR. KNIGHT: Just to prove that there is something to this business of great minds running in similar channels, we have chosen an excerpt from the Perry letter as perhaps being fitting as a way of setting some context for the discussions.

(Slide.)

MR. KNIGHT: Similar from the standpoint of the working folks, so to speak, is the excerpt from the Wolf Creek letter for the -- we hope that during our talk today we could focus to some extent on two things. One is, of course, a discussion of the lower probability, more severe earthquake. The other, however, I think from the standpoint of implementation, it is really the question of needed modifications made to the plant. The logistics, I suppose one might say the practicalities, one might say, of setting about to make plant modifications require that you have some standard that you decide that at some juncture the margins that are available or the situation that exists is not acceptable and you now must go in and make hardware changes. But in order to do that, the designer has to have a standard. He has to know, am I going to have some different spectral intensity? Am I going to have some different spectral shape? All these questions have to be answered.

(Slide.)

MR. KNIGHT: We go a little bit deeper, and we talked about the margins. Are we talking margins on code limits? Would it be within keeping with the philosophy expressed in the letters if we perhaps looked at more advanced analytical techniques that still demonstrated margin-to-failure on some presumably technically founded basis, but distinctly different from what is utilized in the basic design.

If we get into testing of equipment and looking at fragility limits, I can see questions arising as to whether some sort of statistical level would be acceptable. If I tested 17 pieces of identical switch gear to failure, would I take the 84th percentile or would I be looking for 99.9 confidence they would survive at some level? These all become very real questions.

Do you have acceptable fragility if you have what some have already tagged as the SSSE, super-SSE? Would I perhaps be willing to look at something less than the criteria that we apply for functionability? In other words, where I absolutely refuse to accept relay chatter for the design basis, do I accept relay chatter as long as there wasn't structural failure? They are all very reasonable questions.

Last, if I really were talking about very high motions at very high stress levels, would I also depart from such criteria as damping or anything else that has again a technical basis but one which would be different from that which is applied in our usual practice?

MR. OKRENT: Could I offer a comment?

MR. KNIGHT: Yes.

MR. OKRENT: I am speaking for myself, but I have little doubt that the Committee did not intend to suggest that you should strive for the same margins, whether there is a lower probability, more severe earthquake that you strive for for the SSE, and so --

MR. KNIGHT: That is a most significant point to us.

MR. OKRENT: So your point on inelastic, certainly if by including inelastic you can show that various things do well, that would cover those systems and components, in my opinion, and it was couched in terms of risk, you notice.

So where a question is appropriate to answer statistically, that is also, it seems to me, an avenue that one would follow. Not all things may be susceptible to the statistical determination, but some may go that way.

And on the last point, if chatter doesn't affect the ultimate thing you are trying to achieve, then it is all right. If chatter bothers something, there is no unique answer there, I suspect.

MR. KNIGHT: No, absolutely not. And again, as I say, in essence I suppose one might say we finesse the question in the usual applications by simply saying no relay chatter. Therefore you don't get involved in questions about which systems would interact with other systems.

MR. OKRENT: That's right. So if you had a system that could chatter, you would have to show that it is okay.

MR. KNIGHT: Conceivably it might well be an unanswerable question.

MR. OKRENT: Well --

MR. KNIGHT: But again, from our standpoint, these doors are all opened and we've got to be able to provide sufficient guidance to a utility to be able to say, all right, this is an increment, I guess, in our view, an increment above and beyond what was anticipated as being an adequate design basis for this plant, an adequate licensing basis, and now provides sufficient guidance so that their goal is something we can come to grips with.

MR. SIESS: On the question of inelastic, our experience has shown that structures, at least, show a lot more resistance than is easily calculated. You can't always calculate some of those margins that I think are there. But certainly NUREG/CR-0098 is an attempt to formalize, what shall I say a recognition of inelastic behavior.

Wouldn't it be a legitimate guidance to looking at some of those things?

MR. KNIGHT: Well, again, I think it might. Again going to my personal opinion, I certainly think it would be.

MR. SIESS: And if it was actually intended -- well, it was intended, I think, for the SEP plants, but there is nothing wrong with applying those principles to any plant that has already been designed and built, is there?

MR. KNIGHT: I certainly don't believe there is, but as we all know, the experience in the past has not always been the best when the Staff has taken a particular tack and then come before the Committee after the fact. I think it ought to do this job the way it should be done. We really need to develop prior to laying the requirement on a utility the ground rules. There may well be situations that have to be treated that way, but it is in my opinion a very significant effort that could evolve here that would have no foundation for acceptability until we started examining the end product. That is really an unacceptable situation.

MR. SIESS: Well, in some of the previous instances -- I guess North Anna was one, was it not -- what we got back when we asked about margins was simply the margins calculated stress, and those were usually fairly substantial for piping, as I recall, and some other things.

But even if those margins are not substantial, that does not mean we still don't have margin for inelastic behavior which affects both the forcing function and the behavior, right?

MR. KNIGHT: That is certainly true.

If you remember, for instance, taking North Anna, there were some items. One that comes to mind are the hold-down pumps that blew the pumps that were designed to normal design numbers, like 1.01.

I guess I should stop in mid-sentence there. To me we were answering a somewhat different question there, however. There we were saying, all right. This plant, were we to specify seismic motion for this plant today, it would be somewhat greater than that for which it was designed. So let us look at the margins that are available and discern whether or not we would have a satisfactory situation today. And in that context, just saying the margin is 1.0 many would argue is indicative of a totally satisfactory answer. You met your goal, and it would be satisfactory if you designed right up to the line today, and you would have made it.

The more recent letters I believe are asking a significantly different question. That is, regardless of where your SSE falls with regard to our best judgment, our best exercise of technology today, we ought to be able to demonstrate that an event of more severity, albeit less probable, could still be tolerated. It is that somewhat nebulous requirement that I think is going to give us a great deal of difficulty in trying to implement it. You are in the posture again of going to the utility and saying, well you've got to do something better without being able to explicitly tell them what their goal is.

MR. BENDER: I think that might be a misinterpretation, Jim. I think what we have been asking is if it happened, here are the margins, as opposed to saying that you really need to show that margins exist. But there surely is some reserve in these structures, and even though we may not ever expect to use them, knowing that they are there increases our confidence in the event of mistakes and other things.

MR. SIESS: Maybe we need a research project to find out what the seismic margins are.

MR. BENDER: In fact, the Livermore people tried to do it, and so far I haven't seen any good results from it. I saw some very bad use of it yesterday when TVA made their presentation.

MR. KNIGHT: I am not sure you were here in the room or not. I said some of the key words that led us perhaps to think in error, but that any needed modifications be made, given some time period.

MR. OKRENT: Well, let's talk a little bit more about the point because it is sort of central to the whole discussion. It may be that when you look in detail at a specific plant in all aspects with regard to accomplishing safe shutdown -- and that means not getting into a situation where you have a LOCA as well -- in all aspects, given earthquakes of increasingly low probability, the plant can accommodate this. It may in fact have inelastic distortion, and some components may be irretrievably lost that you don't need for this purpose, and so forth and so on.

But just to say that it is designed for some safe shutdown earthquake, if the probability of the safe shutdown earthquake lies in the range of, let's say, one 1000, one to 10,000 per year, that may not be enough if you haven't taken the detailed look.

Now, after you take that detailed look you may decide that the plant in all respects is adequate for less probable earthquakes although there will be nonelastic behavior. Okay.

On the other hand, in some cases you may say it is good in 98 percent but there are some things, based on the existing information, we can't say how it will be 20 years from now when it has aged and so forth. So we are going to need some research or some special testing on similar things in order to see what we can estimate, let's say.

Or it may also be that in some cases you say these things are pretty much going at their limit now, and if we want to have them capable of taking a somewhat less likely earthquake, we need the modification.

Now, up to now, so far as I am aware, there has not been a really systematic look. Things have been sampled. And usually when structures are sampled, if they were seismic class 1 originally, they seem to come out in good shape. I don't recall seeing any. You have a liquefaction question you talk about, they increase it, but aside from that, everything I have seen seems to indicate structures turn out pretty good even without inelastic.

So the guess is on my part that it is other areas. And in fact, you have found that there have been some parts of plants that were supposed to be seismic classified, that the cable trays and so forth weren't so well supported and so forth and so on. But there has not been the systematic look to see that you can with a sufficient degree of assurance get the sufficiently small contribution to risk.

I could foresee an approach to this that didn't start out trying to specify in detail what had to be met for a specific plant. It might be that one approach was to have someone write down how would I do this if I were going to do it? What are the practical ways of trying to include inelastic effects and so forth? What kinds of criteria do I think are appropriate to measure against? What do I expect to be the things for which I have sources of information, and where do I know already that I am limited, and sort of, you might say, lay out a kind of a proposed detailed work scope for the thing for people to look at and reflect upon which would be, from my point of view, a thinking kind of effort.

And it should not start with a detailed prescription.

MR. SHEWMON: Dave, that sounds like it might be a good research program, but I have difficulty seeing it as a licensing criterion.

MR. OKRENT: Well, it is the sort of thing that the people who are expert in the structural seismic area certainly, like, well, some of the consulting companies -- I won't name any single one or two because I might leave one or two out and their feelings would be hurt -- they could in fact lay it out for the structural part. They might need some assistance from people who think about moving parts and electrical systems and so forth, but I think what I am suggesting is the sort of thing that experienced people in the field could propose.

MR. SHEWMON: The experienced people wrote the code, and you said you thought the structures part was pretty good where they looked at it before.

MR. OKRENT: What I said was when they looked at structures that were designed for a certain earthquake and examined how much larger an earthquake would the structures take before they thought there was a reasonable likelihood of failure.

MR. SHEWMON: Not failure going plastic, usually. It is inelastic.

MR. OKRENT: Failure to serve its function.

MR. SHEWMON: Not just that going plastic in some small part. That is not failing to perform its function at all. That is where the margin comes in.

MR. OKRENT: There are different looks people have taken, Paul. I was going to take a different measure, which was failure to serve its function, and then one usually gets a very considerable capability in the structures.

When I see analyses done, it is more than just --

MR. SIESS: The regulations for SSE simply require that they remain functional.

MR. KNIGHT: That's correct, yes.

MR. SIESS: It does not require they stay within any specified allowable stresses at all.

MR. SHEWMON: Jim is beyond this fire drill which he does sort of yearly now, or more frequently, and what we get is ratios of stresses, at least with regard to structural things, and that is usually where it goes plastic, I suspect, or gets up to some yield stress.

MR. KNIGHT: And the reason is one I think of almost pure economics. The information that is available to design houses, if you will, is some calculated stress in accordance with the code, and you want to compare that with the code limit. That type of information you can extract rather readily.

If you are getting into a situation where you are now going to say let's go back and look at this beast on what amounts to a really different basis, and if you want to start with structures, you pretty well have to turn an entirely different group of people loose to sit down and start doing a far more sophisticated analysis. And I am sure that the Staff and the group could argue about the nuances of the methods of analysis for some extended period.

MR. BENDER: The people who were at Livermore last year did an investigation of the design of the plant. They even interpreted it for more severe earthquakes. I really didn't have a great deal of enthusiasm for that program, but having done it, they did develop some methodology for looking at margins.

It seems to me that you might at least look to see whether they learned enough in that to provide some way of making such assessments. I don't disagree with Paul. I think we may be pushing the requirements more than we need to. But nevertheless, if the methodology is there we may as well be prepared to use it.

I am more concerned about the fact that we will find some mistakes that have to be looked at, we'll see an earthquake that is worse than the SSE. But that's my perspective.

MR. KNIGHT: I think, just to play on that for a moment, if I may, I know one of the thoughts often voiced on the Staff is that there seem to be two lines of thought about gaining margin. One is to increase the basic level of the seismic input, and in some discussions they say, well, we found things wrong here and things wrong there, and these things aren't as good as you think they are.

But if that is the problem, then you have to go after the specific problems. You have to have better design quality and control. You have to have better perhaps field inspection, if those things are occurring. And increasing the level of the seismic input for the design basis really isn't going to ask the question.

Just one other point that came to my mind while I was speaking, I know I do it myself, I've probably done it before the Committee a number of times, but I think most of us immediately grapple with the basic structures, because it's pretty straightforward and it's fairly easy to get a handle on and discuss.

In doing such a study, I am more concerned with the situation where a plant has, say, a number of pieces of equipment that have been tested to a certain level. They have that. That's all they have, is that test record.

To go back now and say, well, okay, but what could it really take -- well, I'm by no means demeaning the thought or demeaning the effort. I'm just kind of thinking out loud about the problem. You run into a number of practicalities.

Well, you could argue, you could take similar pieces of equipment. In many cases similar equipment doesn't exist. You can empanel a group of people to look at it and speculate on what it could take, given what it was tested at. I doubt that would be a satisfactory solution.

I think there was an effort to discern after the fact the fragility of this equipment. That is a significant undertaking and it has problems that are quite a bit more severe than, say, going back and re-analyzing a structure, albeit that can be a pretty severe task in itself. But at least it is doable.

I think we can dig out the capability of a piece of equipment to take some more severe -- something more severe than you can discern from what was already done. Clearly, we can look at the actual test that was perform and in most cases discern that the input was in fact more severe than the basic design requirement, just because of the practicalities of testing.

But to go above that I think is a problem that has very significant import from the standpoint of the plant and the money involved. One of the things I intended to finish up with today is that I think we are in an area that starts raising rather significant policy questions.

(Slide.)

By no means do I intend to stand here today and talk about the direct relationship or implications, but there are these things happening. The Commission has spoken on the safety goal. We in the Staff are very properly charged with giving very much deeper consideration to what may or may not be backfit and how that should be handled in accordance with the regulations.

And the question arises, if we are looking at seismic events in excess of those that one develops by meeting the regulations, and we insist that they do meet the regulations when the plant is licensed, that raises the question of whether what we're talking about is even de facto modifications.

MR. OKRENT: Let's look a little bit about this question of meeting the regulations for the moment. Let's think a little bit historically in this regard.

There was a time, I believe, when the Staff and the Applicants and maybe the ACRS looked upon the SSE as being a sufficiently low probability event that if the plant just met it, as it were, you were providing a sufficient level of safety in that regard, met it with not too much margin, and so forth.

I think it is not too hard to go back into some of the memoranda and so forth that were written. The first time the Staff spoke about maybe the SEE through one plant was 10^{-5} per year, and the Committee took enough notice to write a memo to whoever was the Director of Regulation at the time asking him to tell us more about it.

In fact, I just happened, in reading the Grand Gulf construction permit letter in preparation for a Subcommittee meeting coming up not too long after this one, I find that by chance I appended a remark to that CP letter saying I didn't think the SSE for Grand Gulf was a 10^{-6} or 10^{-7} event, which presumably some people were saying back in 1974.

MR. KNIGHT: Right again.

MR. OKRENT: Or I wouldn't have appended the comment.

Now we're not talking about that frequency any more. It is somehow the same SSE. So you could say, well, it's the same regulation. But now the Staff comes in, when they provide a rough estimate they say 10^{-3}, 10^{-4}, and in the next breath they say, but there are margins that give us an acceptably low risk. They may not use that kind of phrase, but that's the implication of it.

In fact, there may well be these margins, but if you haven't really assured yourself everything you need with regard to safe shutdown then you may be relying more on luck than is appropriate.

The point I'm trying to make is, this question of meeting the regulations is a little bit fuzzy. One's knowledge of what the frequency of the SSE, or at least one's opinion on the frequency of the SSE, because I'm not sure it's knowledge, has shifted. And so there is more importance in being assured that the margins exist in everything you need for safe shutdown.

Now, I agree with you, you can't just analyze big structures, things that have been qualified by shaking. There may be some difficult situations, and in some cases it may not even be practical to devise a simulator kind of test.

That doesn't mean, I think, that one shouldn't know that there are these possible situations and let expert opinion look at this and say, yes, I think this is okay, or no, in the same way you looked at things related to fires and made judgments and in some cases made very expensive fixes being required, even though there were differing opinions.

There is nothing truly quantitative in that judgment as to what you must or must not do with regard to a fire. That doesn't prove anything about the other. You've had more than one serious one.

So I think myself that it is relevant to look to understand the problem to see in which areas you are in pretty good shape, in which areas you can examine yourself theoretically, which areas perhaps will lend themselves to related kinds of experiments, or maybe where experiments have been done, and which areas are grey areas, and narrow the things that can be done.

My intuition is there are going to be small pieces here and there that may at least reflect areas of concern, and then maybe by looking at the experiences the Japanese are having on their shakers and so forth, this will tell you you need not be concerned on that one, but you need to go back on another one.

I don't think we know the answer today, but if you don't look you won't know it in three years, either.

MR. KNIGHT: Well, I think clearly, at least from my point of view, as you say, myself when I first joined the Staff and got involved and started asking around, what is the level of likelihood, or whatever word we used at the time, as far as this big earthquake goes, the numbers you mentioned, 10^{-6}, were bandied about.

I guess from our point of view, however, those were secondary to the fact that there was a regulation in place and we met the regulation. I suppose the next step was to go beyond that, perhaps, to the history of that regulation and how it was developed. I am not really competent to speak to that.

Yes, sir?

MR. SHEWMON: We are talking here as if the SSE is an immutable number that indeed somebody, people, can agree on. Yet I've also sat in this meeting and heard a fair discussion about what the SSE should be, and it changes from year to year. And just as with Grand Gulf, Dade can go back and say, well, he didn't think that was good enough. But also with Grand Gulf, as some of the others, it gets increased over the last several years.

Is there anything in the Staff's thinking when they consult their ouija board or their crystal ball, or wherever they get their SSE's, to say that it should increase to be something proportional to this one in 10 or one in 10^4? It seems to me that's at least as good a part of this question as beating on the structure or whatever else.

MR. KNIGHT: I'd like to ask Leon Reiter to comment.

MR. REITER: My name is Leon Reiter, from the Staff.

There really is no such criteria, some number that we look at. I think the kind of numbers that have been presented in the past have been numbers which asked what are estimates of the SSE, these are usually made from a quick survey of the studies. We typically say -- and they typically come out 10^{-3}, 10^{4}. However, this is not to say that this is meant to be some sort of rigorously defined factor which should be used in some calculation to determine adequacy.

MR. SHEWMON: To what extent do you find that the SSE does define adequacy?

MR. REITER: Appendix A. There is no probability in Appendix A. Appendix A refers to the operating basis earthquake.

MR. SHEWMON: I have the other side. The plant must be designed to take an SSE and close down safely. But I'm not getting what criteria the Staff uses in selecting the value of the SSE. Could you briefly help me on that?

MR. REITER: Yes. The SSE is generally defined as an earthquake based on an evaluation of the maximum earthquake potential. With respect to the western United States or areas where there are clearly defined faults or clearly defined structures, then an estimate is made of what we think the estimated maximum potential of those structures are.

MR. SHEWMON: Yes, I've been through that. Go ahead.

MR. REITER: In the eastern United States, where these structures -- we cannot identify these particular structures, then we take what is called the tectonic province approach. There we take large areas of usually similar geometry, and the regulation says we take the largest historical event within that, and we assume that it occurred near the site. No mention is made of probability there.

MR. SHEWMON: So you come up with what you think is the most reasonable maximum earthquake, and another group of experts comes up and says, gee, that will happen at least every 1,000 years, not once every million years; is that right?

MR. REITER: No. What comes up is a discussion as to what is the appropriate tectonic province that one considers. In other words, there may be a discussion as to what the boundaries of that province are.

Very often utilities will define a very small province. We say, no, that's not the appropriate province; we think the larger province would be more appropriate. In doing that, once you increase the size of the province or change the size of the province, then a different earthquake or a larger earthquake will become the controlling earthquake.

There is, however, a stipulation in the appendix that came in as a result of controversy over Seabrook that says, if seismological or geological data warrant it, then we can have an SSE which is larger than the historical earthquake in that particular location if it is felt -- these are the words of the regulation. If it's warranted in that case, we have something larger. In some cases we have that, we have done that.

But there is no specific criteria which links the SSE to some probabilistic number.

MR. SIESS: I was going to say, you are right, there is nothing in the regulations that brings in probabilities, but probabilities have been brought in by us and others as we began to look at PRA. And when you bring the probabilities in, it turns out that it is not very terribly low.

MR. REITER: Dr. Siess, again I think those probabilities -- that we've been asked these questions for years already. I go back to an old discussion we had with Dr. Okrent. The 10^{-5} number as far as I know came in with the Greenwood plant. But since then, in all my six years with the agency, in talking to all the people, I'm not aware of that kind of a number creeping in again.

Again, that number was not-- the numbers 10^{-3} or 10^{-4} were meant to be some sort of rigorous criteria which are applied to some measure of safety. They were meant to say, what is the survey of what some people -- what is opinion, I think Dr. Okrent put it, as to what the SSE, the return period of the SSE would be. That at various times has been applied to the size of the earthquake, density, magnitude, peak acceleration, spectrum, the whole range of things.

So I think the Staff has expressed some concern in attempting to use those rather vague answers in some sort of a rigorous way to define acceptability or lack of acceptability.

MR. SIESS: You'll find the same problem when you try to use PRA in any rigorous way. But let's face it, if I believed that the SSE was a true threshold value and if beyond the SSE something disastrous was going to happen to my plant, and if I take the best evidence I can find, which isn't all that good, that tells me that the probability of exceeding the SSE is as much as one in 1,000, then I think I've got a right to be concerned.

I don't think it is a real threshold value for an awful lot of the plant. I know it isn't for structures, because we have a lot of experience with structures that have been designed for these earthquakes. I can't say the same thing for every single thing in that plant and there are an awful lot of things in that plant that are going to get shook.

Now, the regulations don't help us here. The regulations assumed that the SSE was the maximum earthquake potential, and most of us I think felt in those days that that was the maximum earthquake you could have.

MR. OKRENT: No, I don't think so. And in fact, Leon, I don't have a copy of Appendix A as it was adopted in '73 handy, but my recollection is that for the eastern U.S. the intent was to look at the historical record, and also to allow for the limited history available.

MR. SIESS: Right.

MR. OKRENT: And I think words of that sort were included in Appendix A as adopted.

I would argue that in the phrase, whatever it was, that said the limited history, one could equally well within the context of those words have said, whatever is the historical maximum intensity in my tectonic province, I will add one-quarter of an mmi, I'll add one mmi, I'll add two mmi, to allow for this.

And one could have set up a recipe of this sort and still have been within the words of Appendix A.

MR. SIESS: And you might have been still one in 1,000, Dave.

MR. OKRENT: So I don't think it's anything as was originall written that led one down a clearly defined path to the SSE, even given that we had some way of defining what the provinces are, which is not always easy.

MR. REITER: Excuse me, Dr. Okrent. I think you would be correct if the only statement in Appendix A was what you had said, that we're looking at the eastern United States, it's limited to seismicity. However, there is a very prescriptive part in Appendix A that comes after, which says how you deal with the tectonic province.

Do you take the maximum intensity, maximum earthquake that will occur -- it doesn't say you take it and then you weigh it according to the historical seismicity. It didn't say it at that time.

So I think there was -- I think you're right that the feeling in the past, there was some of this feeling that the eastern United States there is a vagueness limit to historical data, and the general feeling -- and this I get from conversations with people at that time -- was the way to deal with this was to take these large tectonic provinces and assume that the largest earthquake in the large tectonic province is going to occur near the site.

I think that was the attempt to deal with the problem of limited historical data and a lack of knowledge in the eastern United States. Once we had that, it was fairly prescriptive where it would go, except, as I said, in extenuating circumstances and in the Seabrook amendment which was added later.

MR. OKRENT: The words are in the earlier version which permitted one to choose an earthquake larger than what was historical in the province. I think at that time his thinking was that this was a really improbable earthquake compared to 10^{-3} per year.

We have the Greenwood case where 10^{-5} -- there was a little element of shock. So large? That was a shock, not so small.

So it was in that context that this approach was developed and was being used, let's say, in the late sixties and early seventies. And with the change in thinking, and also with the tendency to take this same recipe but move to smaller and smaller provinces, and you take the 100 square miles or 500 square miles in the middle of the state and call that a province, sort of.

We have departed from the basic approach to risk, if I can call it that, that people thought they had in the early seventies, when Appendix A was formulated and adopted, or formulated and second and third drafts were adopted.

MR. KNIGHT: I guess in part what we are seeing here is the interplay or the effect of practice. I think it's something we ought to keep in mind, that regardless, if you will, of the background, in practice, at least in my view, utilities have had every right to presume that if they got their construction permit and some level of -- for seismic design was stipulated, agreed upon, developed, whatever word you want to use, they now had a fixed design basis they could proceed to design to and they would be done.

In some ways I know I was involved in numerous arguments with the utilities myself, when plants would come before you and say, well, if we were to do it today it would be somewhat different. We have often argued that, well, this isn't really backfit; it's just better technology or better understanding of the problems.

MR. SIESS: Jim, how can you make -- you're making a distinction between, let's say, some of the plants we have recently reviewed, where these questions have come up, and some of the older plants, the SEP plants. I'm not sure. Were only those before Appendix A?

MR. KNIGHT: I believe so. Leon has been immersed in that program.

MR. REITER: Certainly phase one and phase two.

MR. SIESS: Those were before Appendix A?

MR. OKRENT: Before Appendix A was adopted there was a draft version of Appendix A.

MR. SIESS: There were six draft versions that went around for as long as the GDC's. We went around with this same kind of a problem on tornadoes. We didn't require any tornado west of the Rocky Mountains, because we thought the probability was so low that we didn't need to worry about it.

And at some point somebody pointed out that the probability of a tornado on the West Coast was one in 1,000 or something like that, which was about the same probability -- the probability was about the same as it was east of the Rocky Mountains. But the tornadoes just couldn't be as big or as strong. And that changed our perception of designing for tornado loadings on the West Coast, and we made a change in the rules, right?

MR. KNIGHT: I guess I have to stop -- there wasn't, if I remember correctly -- and someone please correct me if I'm wrong. Are you using the word "rule" in terms of the regulation, as opposed to the reg guides?

MR. SIESS: Whatever applied. It might be a reg guide, but it's applied like a rule, or a standard review plan or something like that.

MR. KNIGHT: I think tornadoes were covered under GDC.

MR. SIESS: The probability of a safe shutdown earthquake has changed with time once we began to think probabilistically, but we haven't made any changes in the regulations. And that is your problem now. We are trying to get people to look at consequences of an earthquake greater than an SSE, and you are having difficulty finding a regulatory framework in which to do it, right? Is that the essence of the problem?

MR. KNIGHT: Yes, it is, indeed. Actually, there are two. One is the framework and the other is the criteria that we would ask them to meet.

MR. SIESS: Now, NUREG/CR-0098 only applies to structures? I believe it does. It does include some -- it does include some -- no, the inelastic behavior stuff is all based on structures. That was Newmark and Hall, and I can't believe they would have much to say about components. I would include piping.

MR. KNIGHT: That was my hesitation. I was thinking of piping as a structure.

MR. SIESS: Functional survival of electrical components and pumps and valves. But that would certainly be an adequate guideline if you really had to do it, and that doesn't require -- that permits inelastic analysis. It also permits modified spectrum, doesn't it?

MR. KNIGHT: Yes.

MR. SIESS: But as Dave has indicated, and I think we sort of agree, there is not likely to be much problem in the structures. It's more likely to be in the equipment.

MR. KNIGHT: I think if we're going to venture into this type of thing, I would very much hope we would give it enough prior thought to have some uniform approach. And as I said, I freely stand here and admit that the Staff is very much at ends as to what that approach ought to be. And I am most reluctant to charge off and try -- you know, try one approach here and one approach there, and in essence say, wait 'til we see the results and we'll decide whether it's good enough or not.

That's an extremely difficult posture, I think, to put both the Staff and the utility in.

MR. BENDER: Jim, have you looked at the stuff Livermore did last year?

MR. KNIGHT: Certainly. I can't say I personally have been through it all, but I am aware and the Staff has been tracking it.

MR. BENDER: I guess the point I'm trying to make is, if there was any use in that work it was in developing some kind of methodology that might be applied.

MR. OKRENT: Did they include inelastic deformation in their analysis?

MR. BENDER: No, they didn't. But at least they showed how to look at certain kinds of margins.

MR. KNIGHT: We need to go back and look a little harder there.

MR. SIESS: Is this part of the SSMRP?

MR. BENDER: It's in the other part, the load combination.

MR. KNIGHT: Load combination work, right.

MR. SIESS: The Zion PRA looks at the effect of earthquakes on components, but I don't know to what extent their fragilities were based on qualification or whatever.

MR. KNIGHT: My brief look at some of that says they primarily went out and looked under some of the information developed under SSMRP.

MR. SIESS: If I look at Zion, where it takes three times the SSE to start failing components, I get a lot of comfort. But I don't have the slightest idea of how much of that to believe.

MR. OKRENT: It's expert opinion, mostly.

MR. KNIGHT: Mostly. You've got a number of tests that were performed on off-the-shelf equipment, that were performed for some of the original missile projects, and I would think we could probably have a lot of debate about exactly how to apply that.

I personally feel that it's indicative of the performance of a class of equipment, but trying to say that I can now take that information and apply it to component Z at the Perry plant --

MR. BENDER: Well, the least you could do, I would think, would be to do a couple of typical examples to see how something would be treated, if you were concerned about trying to show that you had more margin than was originally in the design, and develop some understanding of what the next thing is you'll take into account.

It's not looking at it as though you had made a design mistake, you hadn't put as much strength in the insulation as you thought. What would you do to accommodate it, other than beef it up? You might be able to show that the next stage of degradation wasn't all that bad.

I'm not sure that you would come to the right conclusion, but if you haven't tried it there's no sense in throwing up your hands.

MR. KNIGHT: I'm certainly not -- on behalf of the Staff, we're not throwing up our hands. We are openly trying to say that we see problems in trying to get on with this recommendation, and we are certainly searching for all the advice we can get as to what was the point of the recommendation and what is viewed as an adequate response, so we can be on target as much as possible.

MR. BENDER: I think I would like -- I'm just speaking for myself at the moment, but the letters you've had so far are not intended, at least in my mind, to have represented a generalized request to look at designs beyond the SSE specified. I think the questions were raised specifically that related to a particular feature of a particular plant.

I don't see any reason to say that it should be a generalized requirement.

MR. SIESS: There are enough plants to be pretty general.

Jim, will a plant like Perry have to do a PRA?

MR. KNIGHT: I think so. Do the folks from Mr. Thadani's shop know offhand?

MR. BUSLIK: Arthur Buslik from the reliability risk assessment branch. I don't really know whether this applies to near-term operating license plants, whether it is required.

MR. SIESS: Let's assume for a minute they would be. Some plants are. Would they have to include seismic in there, or are external events excluded?

MR. OKRENT: It is being done in some and not in others.

MR. SIESS: I asked if it was required, because if they have to do a PRA and they have to do seismic, do they know enough to do that if they don't know enough to do what you think we ask for? And if they do know enough, it seems to me you would satisfy a lot of the questions.

MR. KNIGHT: Maybe to go back to the first of your questions, I don't believe that, at least under current recommendations, that they would be asked to do external events. The question of whether or not they're going to do something notwithstanding, if they do I don't believe it'll include external events at this point. At least current thinking says they would not do external events.

Part of the reason for not doing external events, although the principal reason there, I would say -- and I'll look for Leon to correct me -- is the difficulty in handling the seismological aspects. But certainly, in all the discussions we've had internal to the Staff, the other side of the question -- in other words, how do you get these fragilities, how good are they, how do you use them -- was part and parcel of deciding that the technology really isn't going to apply there.

It's being done, certainly. Some people are certainly worthy of taking a shot at it. How it would be viewed once it's done and how you use it is an entirely different question. I think we're all grappling with that.

MR. SIESS: I guess. Do the levels of ignorance vary that greatly?

MR. KNIGHT: I'm sorry?

MR. SIESS: I wonder if our levels of ignorance vary that greatly.

MR. OKRENT: Could I offer a couple of

comments on what you just said? My impression is that right now in the NREP outline of initiating events, external events that are not included, I think that is a fundamental error that the staff will be making if it stays along that path.

It may have been reasonable when they began IREP. It was already unreasonable by the time they finished IREP, because in fact other groups are including external events. They are turning out to be potentially as important as any others, according to the results more important.

And I think it's not going to be an untenable position for the NRC to take. They may take it, but I think they will find if they stay with it that they will have been wrong.

The other comment I would like to make is, if there's a Commission safety goal which in fact one wants to use to measure for backfitting and one wants to use in considering ATWS or things like that in some suitable way, it is also, it seems to me, the same policy which one has to ask himself about when you talk about seismic.

Again, I think the Commission will find itself at some point in an untenable position if it sort of puts its head in the sand with regard to the seismic issue and says, we have a regulation, and so forth. My own opinion is it's better to have the information to at least know where there may be spots that are not very well known and say, okay, our overall judgment is these are not likely to be too important, than just to retreat to a legalistic one.

I think the legalistic one will become feet of clay for some reason or another at some point.

MR. MARK: Dave, I guess I don't think of what Knight has been describing for us as an attempt to retreat and hide behind the verbiage of legalistic --

MR. OKRENT: I don't want him to be pushed into that.

MR. MARK: But I think I at least would feel it is quite necessary to agree with him that what we have put, and he has shown on the slide, in our letters on Wolf Creek and Perry and possibly Midland 2 is really very vague advice. It was perhaps vague because we didn't exactly know what we thought should be there. It sort of came up in part, if not entirely, from the coming to feel that the SSE was not a highly improbable event. It may be as probable as core melt or something else, and it is proper to take some account of that in some form.

But we didn't write any words which gave a recipe that could be followed in licensing. The needed modifications -- well, needed for what? Well, for a considerably larger earthquake, albeit less frequent. Well, how much larger?

Now, we have not talked, remember, about how we might in any way quantify those words. I think it might be possible, but I don't believe we have done it. If you step up the mmi by one unit, what do you do to the, oh, more or less standard curves of frequency if you go from mmi 8 to mmi 9?

MR. REITER: In the eastern United States, you go into the realm of almost fantasy for the most part. But those particular numbers for the eastern part of the United States are almost in the realm of fantasy. But you're saying if you go up one unit of intensity?

MR. MARK: That was my question.

MR. REITER: Studies in the East indicate that if we double the acceleration -- and that's what happens when we increase the intensity by one --

MR. MARK: One unit of intensity?

MR. REITER: Doubles the acceleration, doubles the intensity. It seems to lead to an increase or a decrease in risk by a factor of five.

MR. MARK: You mean decrease in frequency. You double the acceleration with the unit and you decrease the frequency by a factor of five.

MR. REITER: Right. That is a ballpark estimate. But again, it can't be applied to all ends of the intensity scale.

Dr. Mark, there's something here I feel I must say. I think as a seismologist on the Staff, we all understand the concern about the SSE. I think what concerns us is, one, the use of these vague probabilistic numbers that have been supplied in the past to somehow assess the adequacy of the SSE on the one hand; on the other hand to come up with using numbers like 10^{-5}, 10^{-6}, to describe some future goal which we have to arrive at.

We have enough problems in trying to determine what the level of the SSE is without trying to determine rigorously what a 100,000 million year earthquake is. It is just beyond the state of the art. I think we're perhaps fooling ourselves with these numbers that we gain some safety from them.

MR. MARK: I'm aware of the problems. I'm also aware that if you told me you thought you knew the number exactly for 10^{-5} I wouldn't believe you anyway.

(Laughter.)

MR. MARK: Perhaps that's only because I've heard you talking about the problems.

However, I think that we have some work to do, Dave, as well as suggestions to make, before we can make an operable suggestion, one that somebody could take back to Wolf Creek and say, this is the level to which we ask you to push this.

MR. BENDER: Well, I agree with Carson, really, that we do need more guidance than we have given. And I do not want to set the thoughts I have as specific guidance I would give today, but there are approaches that could be considered.

First of all, some fractions of these plants that we made comments about, if you were to reevaluate the earthquake today you would probably assign a different earthquake than you did when the original licenses were given. That would be one way to decide what level you were trying to address.

Another is to consider a thought we have had many times, that we ought to have some floor on the earthquake that is considered. And some of these plants are designed to a pretty low seismic level. So going up to some specified floor and using that as the basis for finding out what the margin is might be a useful exercise.

I guess the third is to look at this research work that was done and see whether it provides any clue as to how to go about looking at things. I think it would show you where to look. Most pieces of hardware, there's so darn much margin that you could ignore it altogether. There are just a few places where the margin is sufficiently low to make it worth looking at carefully.

I would be inclined to say that, using that work as a reference, you might find some guidance as to what to concentrate on. That is about the best commentary I can make at this stage of the game.

MR. OKRENT: Well, Carson, I would say that the sentence, wherever it is in the Perry letter, we believe it is important that there should be considerable assurance that the combination of seismic design basis and margins and seismic design are such that this accident source represents an acceptable low contribution to overall risk at this plant -- that is a fairly specific kind of general guidance.

It doesn't say how to do it at all, but it does, I think, suggest, at least to me, what one should look for in this regard.

MR. MARK: Look, Dave, I don't disagree with you. I think the way it was phrased in Perry was perhaps the best. The way it was phrased in Wolf Creek didn't come out with that line of thought. It just said, considerably bigger --

MR. OKRENT: Well, as Mike suggested earlier, Wolf Creek was designed with two different design bases, so the Committee was suggesting you might want to take a harder look at the part that had lesser design basis.

MR. MARK: The way it was said in Perry could be harmonized with the present form of the safety goal.

MR. OKRENT: Yes.

MR. MARK: The way it was said in Wolf Creek doesn't accomplish that.

MR. OKRENT: I think the intent was the same, but this particular sentence, I think, could provide, as I say, the general thought.

Can I make a comment? It is conceivable to me that for future plants people will find it useful in trying to meet this objective, if that's your objective, not to have the same design basis seismically speaking for all parts of the plant, because they may know that if I design it for a .2g earthquake for the containment I can handle whatever it is, 1g or something, with no point going to larger numbers, whereas there may be certain actuators or relays or something which, if I have designed it for .2g and shook it for the corresponding, I don't have much margin, at least I don't know I have much margin, so I have to qualify it for something substantially larger or whatever I'm going to do.

In other words, in order to try to meet this kind of objective, it may mean that for the future plant you keep it in mind as you go around. And getting back to the point that Bender made, there may be certain specific areas where things are the most sensitive, so those in fact have some additional either analysis by inelastic to show they're all right or some additional support or whatever, or it may be less support. I don't know which is better, in fact. I'm not trying to enter that argument at all, myself.

MR. BENDER: It's worthwhile to remember that in many cases we have designed it in such a way that we are making the structure too strong. It has been a disadvantage.

MR. OKRENT: It's conceivable.

MR. SIESS: Dave, I certainly agree with the Perry statement, that it represents a goal that is not unreasonable. But it seems to me that the way that is stated the only way you can satisfy it is by doing a PRA, because when we use the words "contribution to risk" to me that conjures up PRA. Now, maybe it doesn't to everybody, but --

MR. OKRENT: Can I comment on that? I think PRA including seismic in principle would be part of doing that. But you might be able to use other PRA's and decide that your plant is similar enough to these others so that you know what the rest of the plant is like, for example. And you might look at seismic under a portion of the PRA for different plants.

But having done a PRA of the kind I've seen, let's say for Zion, I don't think they looked hard enough at the specific plant to provide necessarily the assurance that the actuators and the valves and so forth that you need, and the small lines and so forth, are okay. It was a generic kind of fragility study that was used in Zion, and one has to go back in that area, I think, and do some more thinking.

MR. EBERSOLE: May I ask a point of clarification? To me this thing sort of resolves itself. I realize we've designed LOCA mitigation systems to survive earthquakes, but we've never admitted we're going to have a loss of coolant accident, at least formally. Those strange words, "coincident, not caused by."

If we are going to get with the business of mitigating LOCA's and earthquakes, I think we should be forthright and say that, and be very specific, because life gets a lot tougher when you say that. You've got a lot more things to worry about than a simple reactor trip and the shutdown cooling function. And I'd like to see some policy statements in fact clarifying very well indeed that that's the way we're going to go.

Then we're going to have to look at a much larger field of seismic margins.

MR. MARK: Competence of equipment to cope with the loss of coolant accident coincident with the following earthquake. If, on the other hand, we deny this combination, then we've got a much smaller field of problems to work with, and I think a much better chance of showing we can do it without gross costs involved.

As a matter of fact, we just heard yesterday, it turns out in their seismic analysis, unlike the -- one of the earlier plants, where they had pinion type pumps that turned out to be weak, that by some simple braces that prevents the swinging of these things apparently they have fixed that. We're going to look at it again. Their problem was based on relay chatter and pump performance.

This is a housekeeping problem that may get into fine detail. We may find two dollar items right at the plant we didn't look at that really are Achilles' heels, in spite of our heavy investments in heavy structures and physical restraints and piping systems and so forth, that some of the little gingerbread in the design just won't work.

MR. KNIGHT: I certainly agree that as a result of such effort you find Achilles' heels, weak links, that the effort has paid off handsomely.

You bring up another point, though. It gets back to this business of the nitty gritty, I suppose one might say, of regulation. You end up chasing your tail a little bit.

As you say, if I find that the first time around there were some really weak link and now the next weak link is, you really need some point where you decide, okay, there certainly is always going to be something which is a weak link, so to speak, but its capacity is so much larger than whatever the goal is that we are right.

MR. EBERSOLE: I think there may be a general recipe that the weakest link ought to be the costliest link, that you shouldn't wind up with some poorly designed, $2 items.

MR. KNIGHT: That is certainly fundamental cost-benefit.

MR. BENDER: There are only 15 members of this Committee, so there are only 15 opinions.

(General laughter.)

MR. BENDER: But I think that a pretty good approach is to try to compare the hardware that exists in these plants that we have asked about, the hardware in the plants that have the higher seismic design requirement than the one that we are talking about, and seeing what the significant differences are, and that at least would give you some feeling for the problem.

I guess I am not in a position to say how much of that needs to be done, but it will turn out that most of the equipment is about of the same class. I would be much more concerned about trying to jack up the seismic requirements at Diablo Canyon from what it is now than I am from trying to suggest that you take another look at Summer, if I can use the extreme.

While I don't like to ask you to do more analysis, I don't think the problem is all that big.

MR. OKRENT: I think Mike's point about trying to see what was done at some of these higher design basis plants, that could be an interesting way.

We are going to have to break in a few minutes because I think there is another meeting that begins at 2:30.

I would like to suggest that maybe we get together again for an informal discussion in a couple of months or something. Maybe we'll have developed some possible approaches or whatever, and maybe somebody can do a little looking one way or another as to --

MR. KNIGHT: I think that would be most useful. We will certainly be looking at how to tackle this problem.

As I said, I am convinced myself that there are some significant policy questions here. We will also be trying to come to grips with those. I just don't want to be in a posture of saying yes, we will come back and lay out a plan for you.

MR. SIESS: Dave, why don't we have a meeting with SSMRP and instead of listening to them telling us what they have been doing, we can ask them some questions about seismic margins and see if they can answer them. I learned more about the project than what we were in the previous meeting.

MR. OKRENT: We have already had a suggestion for how we might have some future meetings.

I might just say as a point of information that at the LMFBR safety meeting in Lyons, the issue that stuck out in my mind is the one that the French who were the furthest along emphasized, that for seismic design they seem to be thinking in terms of possibly putting one of their isolation devices under future LMFBRs to facilitate seismic design.

MR. SIESS: That will help the expert market in California.

MR. OKRENT: It is interesting.

MR. EBERSOLE: Is that because they do not want to admit TVA?

MR. OKRENT: No. You have to keep certain functions going. They have a thin wall system, you have sloshing possibly, and either because it -- I just wanted to know if you were focusing on that.

MR. KNIGHT: Is that published?

MR. OKRENT: There will be a proceeding on one of the papers given by one of the French engineers.

MR. SHEWMON: What stuck out as most important in Dave's mind will be published only in the minutes of the meeting.

MR. OKRENT: Well, I guess we had better then thank the Staff for coming down and talking about this, and we will adjourn then this subcommittee meeting.

(Whereupon, at 2:25 o'clock p.m. the meeting of the subcommittee was adjourned.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

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in the matter of: ACRS/Subcommittee on Extreme External Phenomena

Date of Proceeding: August 11, 1982

Docket Number: ____________________

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Jane N. Beach

Official Reporter (Typed)

Jane N. Beach

Official Reporter (Signature)

THE SITE-SPECIFIC PORTIONS OF THE PLANT, INCLUDING VITAL ASPECTS OF THE ULTIMATE HEAT SINK AND ASSOCIATED SYSTEMS, WERE DESIGNED FOR A 0.12 g EARTHQUAKE, AND ARE BEING REANALYZED FOR AN EARTHQUAKE REPRESENTED BY SITE-SPECIFIC RESPONSE SPECTRA THAT ARE ENCOMPASSED BY REGULATORY GUIDE 1.60 SPECTRA ANCHORED AT A ZERO-PERIOD ACCELERATION OF 0.15 g. THE STANDARD PORTION OF THE PLANT, ON THE OTHER HAND, WAS DESIGNED FOR A 0.20 g EARTHQUAKE WITH THE USUAL MARGINS OF SAFETY AND THUS WOULD BE EXPECTED TO WITHSTAND A CONSIDERABLY LARGER EARTHQUAKE WITHOUT FAILING IN SUCH A MANNER AS TO CAUSE A SEVERE ACCIDENT.

WE DO NOT HAVE CONFIDENCE THAT ALL VITAL ASPECTS OF THE ULTIMATE HEAT SINK AND ASSOCIATED SYSTEMS HAVE MARGINS SUFFICIENT TO PROVIDE AN APPROPRIATE LEVEL OF RESISTANCE TO A LOWER PROBABILITY, MORE SEVERE EARTHQUAKE. WE RECOMMEND THEREFORE THAT THE SEISMIC MARGINS INHERENT IN THE COMPONENTS OF THE ULTIMATE HEAT SINK AND ASSOCIATED SYSTEMS BE INVESTIGATED FURTHER AND THAT ANY NEEDED MODIFICATIONS BE MADE BEFORE THE PLANT RESUMES OPERATION AFTER THE SECOND REFUELING.

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WE RECOMMEND THAT THE APPLICANT AND THE NRC STAFF CONDUCT STUDIES TO EVALUATE THE MARGINS AVAILABLE TO ACCOMPLISH SAFE SHUTDOWN, INCLUDING LONG-TERM HEAT REMOVAL, FOLLOWING AN EARTHQUAKE OF SOMEWHAT GREATER SEVERITY AND LOWER LIKELIHOOD THAN THE SAFE SHUTDOWN EARTHQUAKE. WE BELIEVE IT IS IMPORTANT THAT THERE SHOULD BE CONSIDERABLE ASSURANCE THAT THE COMBINATION OF SEISMIC DESIGN BASIS AND MARGINS IN THE SEISMIC DESIGN IS SUCH THAT THIS ACCIDENT SOURCE REPRESENTS AN ACCEPTABLY LOW CONTRIBUTION TO THE OVERALL RISK FROM THIS PLANT. WE RECOMMEND THAT ANY NEEDED MODIFICATIONS BE MADE BEFORE THE PLANT RESUMES OPERATION FOLLOWING THE SECOND REFUELING. WE WISH TO BE KEPT INFORMED ON THE PROGRESS AND RESULTS OF THESE STUDIES.

STANDARDS FOR SEISMIC EVENT

- o 10 CFR 100 APPENDIX A
- o SSE + 10% ?
- o SSE + 100% ?

STANDARDS FOR PLANT MODIFICATION

- o MARGINS ON CODE LIMITS ?
- o INELASTIC ANALYSES ?
- o STATISTICAL LIMITS FOR TESTED EQUIPMENT
- o ACCEPTABLE FRAGILITY

  RELAY CHATTER VIZ. STRUCTURAL FAILURE
- o INCREASED DAMPING ETC.

SIGNIFICANT POLICY QUESTIONS

- COMMISSION SAFETY GOAL
- BACKFIT
- MODIFICATION TO REGULATIONS

Attachment B

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 13, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE PERRY NUCLEAR POWER PLANT, UNIT 1

During its 267th meeting, July 8-10, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of the Cleveland Electric Illuminating Company (Applicant), acting on behalf of itself and as agent for Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and the Toledo Edison Company, for a license to operate the Perry Nuclear Power Plant, Units 1 and 2. The plant is to be operated by the Cleveland Electric Illuminating Company. A tour of the facilities was made by members of the Subcommittee on the morning of June 28, 1982, and a Subcommittee meeting was held in Cleveland, Ohio on June 28 and 29, 1982 to consider the application. During its review the Committee had the benefit of discussion with representatives of the Applicant, the NRC Staff, and members of the public. The Committee also had the benefit of the documents listed. The Committee commented on the application for a permit to construct this plant in its reports dated December 12, 1974 and May 12, 1975.

The Perry Nuclear Power Plant is located in Lake County, Ohio near Lake Erie approximately 35 miles northeast of Cleveland, Ohio and 21 miles southwest of Ashtabula, Ohio. Units 1 and 2 use General Electric BWR-6 nuclear steam supply systems with a rated power of 3579 MWt and a Mark III pressure suppression containment system with a design pressure of 15 psig. Construction of Unit 1 is about 83% complete and Unit 2 is about 43% complete.

Because loading of fuel for Unit 2 is scheduled for May 1987, the Committee does not believe it appropriate to report at this time on the operation of Unit 2.

Our review included the management organization, technical support staff, status of operational staffing, and the training program. This is the first nuclear power plant to be operated by the Applicant. The plant staff has a minimum amount of boiling water reactor (BWR) nuclear background. We agree with the NRC Staff on the urgent need for additional personnel with BWR experience within the operating management. The Applicant should fill the position of Superintendent of Plant Operations in the near future. Experienced senior technical support personnel should be included in the staffing plans of the Applicant. This matter should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

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As a result of adverse experience on the Perry project several years ago, the Applicant restructured its quality assurance procedures and its quality control and assurance organization. The revised organization has been reviewed and audited by the NRC Staff. We wish to receive a report from the NRC Staff which discusses design and construction problems, their disposition, and the overall effectiveness of the effort to assure appropriate quality.

The Applicant has committed several technical staff members to matters related to probabilistic analysis and studies of systems interactions. We believe that efforts of this sort by the operating utilities are to be encouraged.

The Mark III suppression pool dynamic loads have been identified as an Outstanding Issue in the NRC Staff's review. The NRC Staff has provided the Applicant with a proposal for the appropriate design basis loads, and it appears that the Perry design will be able to accommodate these loads. Additional concerns with the design of the Mark III containment have been recently brought to our attention. The NRC Staff is currently assessing these issues for impact on the Mark III design. We will continue to discuss with the NRC Staff, on a generic basis, Mark III suppression pool dynamic loads and other additional Mark III issues.

Hydrogen control systems for Mark III containments are being developed by the Mark III Owners Group. Efforts by this Owners Group are being directed toward the development of a hydrogen ignition system which makes use of distributed ignition sources. The NRC Staff has indicated that they will be able to meet with the Committee on this matter in the near future. We expect to review this system on a generic basis. Acceptability of this system is designated as a License Condition.

We recommend that the Applicant and the NRC Staff conduct studies to evaluate the margins available to accomplish safe shutdown, including long-term heat removal, following an earthquake of somewhat greater severity and lower likelihood than the safe shutdown earthquake. We believe it is important that there should be considerable assurance that the combination of seismic design basis and margins in the seismic design is such that this accident source represents an acceptably low contribution to the overall risk from this plant. We recommend that any needed modifications be made before the plant resumes operation following the second refueling. We wish to be kept informed on the progress and results of these studies.

During our review, the NRC Staff identified a number of other License Conditions, Confirmatory Matters, and Outstanding Issues which remain to be resolved. Except for the issue of turbine missiles, we are satisfied with the progress on these topics, and we believe that they should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed concerning resolution of the turbine missile issue, and wish to receive a technical report which discusses and evaluates the problems involved.

If due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, the ACRS believes there is reasonable assurance that the Perry Nuclear Power Plant, Unit 1 can be operated at power levels up to 3579 MWt without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References

1. Cleveland Electric Illuminating Company, "Final Safety Analysis Report, Perry Nuclear Power Plant, Units 1 and 2," with Amendments 1-6
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report, Perry Nuclear Power Plant, Units 1 and 2," USNRC Report NUREG-0887, dated May 1982
3. Memorandum from D. Houston/J. Kudrick, NRC, to A. Schwencer/W. Butler, NRC, Subject: Summary of May 13, 1982 telecon with John Humphrey - Concerns about Grand Gulf Mark III Containment, dated May 18, 1982
4. Letter from John M. Humphrey, Humphrey Engineering, Inc., to L. F. Dale, Mississippi Power and Light, Subject: BWR-6/Mark III Containment Design Issues, dated May 8, 1982

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 13, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SUITABILITY OF THE CLINCH RIVER BREEDER REACTOR PLANT SITE

During its 267th meeting, July 8-10, 1982, the ACRS reviewed NUREG-0786, "Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant" and considered the suitability of the proposed site for such a plant. The matter was also discussed on June 24, 1982 at a joint meeting of the Subcommittees on Clinch River Breeder Reactor and Site Evaluation. During both meetings, we had the benefit of input from representatives of the NRC Staff and the Department of Energy (Applicant). We also had the benefit of the documents listed below, as well as a direct discussion with the author of Reference 4.

The proposed Clinch River Breeder Reactor (CRBR) plant site is located in Roane County in east-central Tennessee, approximately 25 miles west of Knoxville and within the city limits of Oak Ridge, Tennessee. The site consists of approximately 1,364 land acres on a peninsula formed by a meander in the Clinch River. It is bounded on three sides by the River and on the north by the Department of Energy's (DOE) Oak Ridge Reservation. The site property is owned by the Federal Government, and the portions of the site required for constructing and operating the plant will fall under the custody of DOE.

The CRBR plant will be a single-unit electric power plant with a liquid sodium-cooled loop-type breeder reactor utilizing a fuel of mixed uranium-plutonium oxides. With the initial reactor core, the design power will be 975 MWt, and the net output will be 350 MWe.

DOE has requested a Limited Work Authorization (LWA-1) to begin nonsafety-related site preparation activities. It is required in 10 CFR 50.10 that, before an LWA-1 can be granted, an Atomic Safety and Licensing Board (ASLB) must determine that there is reasonable assurance that the proposed site is a suitable location from a radiological health and safety standpoint for a nuclear power reactor of the general size and type proposed. Our review was made in response to an NRC Staff request in connection with the required ASLB determination. The NRC Staff carefully defined the scope of the review to consider whether the site is suitable for a reactor "of the general size and type" of the CRBR; the current design of the CRBR plant itself was not evaluated.

Among the topics considered in this review were the location and distribution of population around the site; the geology, seismology, and hydrology of the site; an assumed Site Suitability Source Term; and the risks to be expected from a plant of the CRBR type.

As part of its approach to trying to make the risks from an LMFBR comparable with those from a light water reactor (LWR), the NRC Staff provided review criteria for CRBR core disruptive accidents. We believe that this appears to be a reasonable first approach but also believe that at the construction permit stage substantive assurance will be needed that such criteria are being met. We wish to note that we do not necessarily agree with all the LMFBR Design Criteria specified in Appendix A of NUREG-0786.

The NRC Staff appears to have accepted the Applicant's assertion that a CRBR type plant would not represent an undue hazard to the K-25 Plant. We recommend that the Staff confirm through an independent assessment that the potential effects of a CRBR type plant on the K-25 plant are acceptable.

With regard to the seismic design of this plant, we believe it is important that the combination of seismic design basis and margins in the seismic design be such that this accident source represents an acceptably low contribution to the overall risk from the plant. We believe this matter will warrant detailed examination at the construction permit stage to assure that necessary margins are available for all important systems and components.

The NRC Staff has concluded that the CRBR plant can be designed and constructed in such a manner that it will present no greater risk to the health and safety of the public than an LWR plant meeting current safety criteria. We believe that the proposed site is suitable for such a plant.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References

1. U.S. Nuclear Regulatory Commission, "Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant," NUREG-0786, dated June 1982, Revision to March 4, 1977 Report
2. Letter from J. R. Longenecker, DOE, to P. Boehnert, ACRS, concerning earthquake recursion relationships, dated July 7, 1982
3. Handout from NRC Staff (undated) titled, "Review Criteria for CRBR Core Disruptive Accidents"
4. Letter from T. B. Cochran, National Resources Defense Council to P. Shewmon, ACRS, dated July 7, 1982

2P:3580

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 8, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS INTERIM REPORT ON MIDLAND PLANT, UNITS 1 AND 2

During its 266th meeting, June 3-5, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of Consumers Power Company for a license to operate the Midland Plant, Units 1 and 2. This application was also considered at Subcommittee meetings held on April 29, 1982 in Washington, D. C., on May 20-21, 1982 in Midland, Michigan and on June 2, 1982 in Washington, D. C. On May 20, 1982 members of the Subcommittee toured the plant. In the course of these meetings the Committee had the benefit of discussions with representatives and consultants of Consumers Power Company, Babcock and Wilcox Company, Bechtel Corporation, the Nuclear Regulatory Commission Staff, and members of the public. The Committee also had the benefit of the documents listed below.

The ACRS reported on June 18, 1970 regarding the construction permit application for the Midland Plant; on September 23, 1970 regarding several amendments to the application; and on November 18, 1976 regarding applicable generic matters.

The Midland Plant site is located on the south bank of the Tittabawassee River adjacent to the southern city limits of Midland. The main industrial complex of the Dow Chemical Company lies within the city limits directly across the river from the site. There are about 2000 industrial workers within one mile of the site, and the estimated 1980 population was about 51,400 residents within five miles of the site. This makes the Midland site one of the more densely populated sites at distances close to the Plant.

Each of the two Midland units employs a Babcock and Wilcox designed nuclear steam supply system rated at 2468 MWt with a stretch power rating of 2552 MWt. The Midland Plant is unique in that the heat generated will be used not only to produce electricity but also to produce process steam for the Dow Chemical Company plant via a tertiary system.

The Midland Plant has been the subject of several major problems related to quality assurance during plant construction. One of these problems relates to the soil fill under several safety-related structures. The deficiencies relating to soil fill have led to excessive settlement and some cracking of these structures, and have also introduced questions concerning the adequacy of protection against liquefaction of the granular portions of the fill in the event of strong vibratory motion accompanying an earthquake.

The Applicant has proposed and is implementing, under close surveillance by the NRC Staff, remedial measures with regard to the foundation deficiencies. We are generally satisfied with the approach being taken, subject to confirmation of the overall quality assurance program and the seismic design basis. Both of these items are discussed below.

With regard to quality control of design and construction, the report of the NRC Staff's Systematic Assessment of Licensee Performance (SALP) review for the period July 1, 1980 to June 30, 1981 revealed deficiencies in the installation of piping and piping suspension systems, in the pulling of electrical cables, and in the handling of problems relating to soils and foundation. Deficiencies by the Applicant in the handling of soils-related matters have continued to occur, subsequent to issuance of the SALP report. We believe that the NRC Staff is handling the corrective actions for specifically identified quality assurance deficiencies in an appropriate manner.

In view of the overall concern about Midland quality assurance the NRC should arrange for a broader assessment of Midland's design adequacy and construction quality with emphasis on installed electrical, control, and mechanical equipment as well as piping and foundations. We wish to receive a report which discusses design and construction problems, their disposition, and the overall effectiveness of the effort to assure appropriate quality.

Our reservation concerning seismic design relates to the lack of adequate assurance that the Midland Plant will be capable of accomplishing shutdown heat removal for low probability earthquakes more severe than the safe shutdown earthquake (SSE). The Midland seismic design basis at the construction permit stage corresponded to a MMI VI, peak ground acceleration of 0.12g, employing a modified Housner spectrum. For the operating license review, the NRC Staff has reevaluated the original seismic design basis and the Applicant and the NRC Staff have agreed on the use of site-specific analyses which have led to increases in the design response spectra for frequencies above about 2 cycles/sec.

Historically, no earthquakes stronger than the newly proposed SSE have occurred within 200 miles of the Plant. However, expert opinion differs widely on the exceedance frequency of the proposed SSE and on the severity at the site of earthquakes whose likelihood is less than 1 in 10^4 or 1 in 10^5 per year.

The Applicant is currently reevaluating by selective audit the seismic capability of the plant, as originally designed, to withstand the revised SSE. Measures taken to assure safe shutdown in the event of an earthquake include the use of dewatering to reduce the potential for soil liquefaction. We recommend that all systems and components important to decay heat removal be carefully evaluated for their ability to accomplish necessary functions in the unlikely event of lower-probability, more severe earthquakes in order to provide the necessary degree of assurance. This matter should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed about the resolution of this matter. We believe that any recommendations for changes in the plant resulting from this evaluation should be implemented by the end of the second refueling outage.

The Applicant has agreed to provide core exit thermocouples, a hot-leg-level measurement system, and subcooled margin monitors as instrumentation to detect inadequate core cooling. Consumers Power Company also plans to include a remotely operable vent on top of both inlet loops to the steam generators; however, Consumers has not committed to supply a high point vent on the reactor vessel head. This matter should be resolved in a manner satisfactory to the NRC Staff. The ACRS recommends that the Applicant review further the potential for providing indications of water content or level within the reactor vessel.

The staff of the Applicant includes many personnel who have had nuclear power plant experience. However, operating experience with this B&W type power reactor is limited, and the NRC Staff is requiring that at least one person having experience on a large commercial PWR be included on each shift for one year. We support the NRC Staff position.

The Applicant's experience with the operation of nuclear power plants should, in principle, place Consumers in a favorable position to provide continuing, careful oversight of the operations at the Midland Plant. In view of some prior adverse operating experience at the Palisades Plant however, we recommend that the NRC Staff institute an augmented audit of operations at Midland, at least during the early years of operation at power.

We have reviewed the evaluation made of the tertiary process steam system for use by Dow Chemical Company. This system appears not to impose any unacceptable impacts either on the safe operation of the Midland Plant or on the people working at the Dow Chemical Company.

The Applicant has undertaken an effort to have a probabilistic risk assessment (PRA) performed for the Midland Plant and stated that the results will be available in the fall of 1982. We believe it desirable to have plant-specific PRAs performed for each commercial nuclear power plant and that it is particularly appropriate for the Midland Plant because of its relatively high, close-in population density. We wish to have the opportunity to review the Midland PRA with assistance from the NRC Staff, and to offer comments or recommendations as appropriate. We do not believe that this review need delay licensing of the Midland Plant for operation.

Recently, questions have come to light in connection with B&W plants concerning the availability of natural circulation in the presence of an interrupted or continuing small break loss-of-coolant accident. We wish to see a proposed NRC Staff resolution of this issue.

The Applicant described an extensive systems interactions study being undertaken for the Midland Plant. We wish to be informed of the results of this study.

We believe that, in view of the population density near this plant, additional prudence is appropriate for the Midland Plant in the resolution of the ATWS issue and other Unresolved Safety Issues.

We endorse the participation of Dow Chemical Company plant personnel in emergency procedures developed on the basis of an assumed failure at the Midland Plant. Similarly, there should be active participation by Midland Plant personnel in emergency procedures developed on the basis of an assumed failure at the Dow Chemical plant. The Applicant and the NRC Staff should promote continued coordination of these types of relationships, as well as those involving appropriate state and local groups to assure that the capability for an effective emergency response is developed and maintained.

With regard to the eleven items identified in the ACRS Supplemental Report on Midland Plant, Units 1 and 2 dated November 18, 1976, we have the following comments. The issues related to vibration and loose-parts monitoring, potential for axial xenon oscillations, behavior of core-barrel check valves during normal operation, fuel handling accidents, effects of blowdown forces on core internals, LOCA-related fuel rod failures, and improved quality assurance and in-service inspection for the primary system have all been resolved or are in a confirmatory stage of being resolved. Separation of protection and control equipment has been accomplished in an appropriate manner; however, the safety implications of control systems remains an Unresolved Safety Issue directly applicable to Midland. Resolution awaits completion of the NRC Staff Task Action Plan A-47. The effect of ECCS induced thermal shock on pressure vessel integrity has been resolved in part; however, the Unresolved Safety Issue on pressurized thermal shock will apply. Environmental qualification of equipment remains a generic issue which is under review by the NRC Staff and whose resolution will apply to the Midland Plant. Instrumentation to follow the course of an accident has been resolved in part by the development of revised Regulatory Guide 1.97. We do not believe that licensing of the Midland Plant for operation need await further resolution of any of the eleven issues discussed above.

The various other matters identified by the NRC Staff as open or confirmatory in the Safety Evaluation Report should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept advised concerning resolution of the turbine missile issue.

The ACRS believes that, subject to satisfactory completion of construction and staffing and if due regard is given to the comments above, the Midland Plant, Units 1 and 2 can be operated at power levels up to 5 percent of full power with reasonable assurance that there is no undue risk to the health and safety of the public.

We defer our recommendation regarding operation at full power until we have had the opportunity to review the plan for an audit of plant quality and the proposed resolution of the question regarding natural circulation in the presence of a small break LOCA.

Dr. Kerr did not participate in the Committee's review of this matter.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. Consumers Power Company, "Midland Plant Units 1 and 2 - Final Safety Analysis Report" including Amendments 1-43
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Midland Plant, Units 1 and 2," NUREG-0793, dated May 1982
3. U.S. Nuclear Regulatory Commission, "NRC Licensee Assessments," NUREG-0834, dated August 1981
4. Letter from J. Cook, Consumers Power Company, to J. Keppler, NRC, Subject: Midland Project Response to Draft SALP Report, dated May 17, 1982
5. Letter from J. Cook, Consumers Power Company, to J. Keppler, NRC, Subject: Midland Project Quality Assurance Program Update, dated April 30, 1981

6. Letter from J. Hind, NRC, to J. Cook, Consumers Power Company, Subject: Systematic Assessment of Licensee Performance (SALP), dated April 20, 1982
7. Letter from J. Cook, Consumers Power Company, to H. Denton, NRC, Subject: Summary of Soils-Related Issues at the Midland Nuclear Plant, dated April 19, 1982
8. Letter from K. Drehobl, Consumers Power Company, to D. Fischer, ACRS, Subject: Midland Project Soils Information, dated April 12, 1982
9. Statement of Ms. M. Sinclair to ACRS, dated June 4, 1982
10. Letter from B. Stamiris to Dr. D. Okrent and ACRS Members, Subject: Midland OL Review, dated May 29, 1982
11. Letter from M. Sinclair to Dr. P. Shewmon, ACRS, Subject: Midland OL Review, dated May 28, 1982
12. Statement by Dr. C. Anderson to ACRS Midland Plant Subcommittee dated May 20-21, 1982
13. Statement by Ms. M. Sinclair to ACRS Midland Plant Subcommittee dated May 20-21, 1982
14. Letter from B. Stamiris to D. Fischer and ACRS Members, Subject: Soil Settlement and QA Issues, dated May 20, 1982
15. Letter from M. Sinclair to Dr. C. Siess, ACRS, Subject: Midland Soil Settlement, dated April 26, 1982

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 11, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE WOLF CREEK GENERATING STATION, UNIT NO. 1

During its 265th meeting, May 6-8, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of Kansas Gas and Electric Company (KG&E), Kansas City Power and Light Co. and Kansas Electric Power Cooperative, Inc. (Applicants) for a license to operate the Wolf Creek Generating Station, Unit No. 1. The Station is to be operated by KG&E. A Subcommittee meeting was held in Emporia, Kansas, on April 21-22, 1982, to consider this project. A tour of the facility was made by members of the Subcommittee on April 21, 1982. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, Westinghouse Electric Corporation, Bechtel Power Corporation, the Nuclear Regulatory Commission (NRC) Staff, and with members of the public. The Committee also had the benefit of the documents listed below. The Committee commented on the construction permit application for this plant in its report dated October 16, 1975.

The Wolf Creek Generating Station is located in Hampdon Township, Coffey County, Kansas. The site is in eastern Kansas approximately 53 miles south of Topeka, and 100 miles east-northeast of Wichita. The nearest population center is Emporia, Kansas, 28 miles west-northwest of the site (estimated 1980 population of 25,019).

The Wolf Creek Generating Station will be the first commercial nuclear power plant in the state of Kansas. It should be assured that state and local agencies are qualified to respond to possible emergency situations associated with the operation of the Wolf Creek Generating Station.

The Station will use a Westinghouse, four-loop, pressurized water, nuclear steam supply system having a rated power level of 3425 MWt. Unit 1 employs a cylindrical, steel-lined, reinforced, post-tensioned concrete containment structure with a free volume of 2.5 million cubic feet. The Wolf Creek Generating Station uses the Standardized Nuclear Unit Power Plant System (SNUPPS) design. It is one of two plants built to this design. The Committee reported on the operating license application of the other plant (Callaway Plant Unit No. 1) in its November 17, 1981 report to you.

The Wolf Creek Generating Station is the first nuclear power plant to be operated by KG&E. The Committee reviewed KG&E's management organization, experience, and training programs. We were favorably impressed by the general competence and attitude of KG&E's personnel. Nevertheless, we wish to emphasize the importance of KG&E's building a strong in-house capability for analyzing and understanding the nuclear-thermal-hydraulic behavior and systems performance of this plant.

To strengthen the shift structure during the initial period of operation, KG&E plans to augment each shift with a consultant who is an experienced, previously licensed PWR operator. These consultants will serve for a period of one year after startup. In addition, KG&E has retained the services of a consultant with considerable commercial nuclear experience to act as a technical assistant to the Plant Superintendent through the initial loading of fuel. We believe the technical assistant to the Plant Superintendent and the "experienced operator consultants" should be retained until the operating organization has developed an experience base involving those operational duties of importance to public safety. This experience base should be defined by the NRC Staff in consultation with operational experts and incorporated into the regulatory requirements instead of using arbitrary operating time periods as a basis for measuring skill. We encourage the practice of assigning the Senior Reactor Operator (SRO) candidates to extended tours of service at operating nuclear power plants, and recommend that others in the operations staff participate in such a program to the extent practical.

KG&E has proposed, as an alternative to a Shift Technical Advisor (STA), that at least one SRO on each shift have the training and background required for an STA. This approach appears to us to meet the need which originally led to the requirement of an STA. However, it is not clear that the level of training given to the SROs will correspond to that intended for STAs, and we recommend that the Staff review this matter carefully.

The site-specific portions of the plant, including vital aspects of the ultimate heat sink and associated systems, were designed for a 0.12 g earthquake, and are being reanalyzed for an earthquake represented by site-specific response spectra that are encompassed by Regulatory Guide 1.60 spectra anchored at a zero-period acceleration of 0.15 g. The standard portion of the plant, on the other hand, was designed for a 0.20 g earthquake with the usual margins of safety and thus would be expected to withstand a considerably larger earthquake without failing in such a manner as to cause a severe accident.

We do not have confidence that all vital aspects of the ultimate heat sink and associated systems have margins sufficient to provide an appropriate level of resistance to a lower probability, more severe earthquake. We recommend therefore that the seismic margins inherent in the components of the ultimate heat sink and associated systems be investigated further and that any needed modifications be made before the plant resumes operation after the second refueling.

Other issues have been identified as Outstanding Issues, License Conditions, and Confirmatory Issues in the Staff's Safety Evaluation Report dated April 1982; these include some TMI Action Plan requirements. Except as noted above, we believe these issues can be resolved in a manner satisfactory to the NRC Staff and recommend that this be done.

We believe that, if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, training, and preoperational testing, there is reasonable assurance that the Wolf Creek Generating Station, Unit No. 1 can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:
1. "Final Safety Analysis Report for Standardized Nuclear Unit Power Plant System," with Revisions 1-8.
2. "Final Safety Analysis Report, Wolf Creek Generating Station Unit No. 1," with Revisions 1-8.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station, Unit No. 1," NUREG-0881, dated April 1982.
4. Written statement by John M. Simpson, Attorney for Intervenors, Re: Emergency Planning Procedures and Plans - Wolf Creek Plant, dated April 22, 1982.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 9, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON CLINTON POWER STATION UNIT 1

Dear Dr. Palladino:

During its 263rd meeting, March 4-6, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of the Illinois Power Company, the Soyland Power Cooperative, Inc., and the Western Illinois Power Cooperative, Inc. (Applicant) for a license to operate the Clinton Power Station Unit 1. The plant is to be operated by the Illinois Power Company. A tour of the facility was made by members of the Subcommittee on the morning of February 25, 1982 and a Subcommittee meeting was held in Decatur, Illinois on February 25-26, 1982 to consider this application. During its review the Committee had the benefit of discussion with representatives of the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee commented on the application for a permit to construct this Station in its report dated April 8, 1975.

The Clinton Power Station is located in DeWitt County in east-central Illinois about 6 miles east of the city of Clinton and 22 miles north-northeast of Decatur. Unit 1 uses a General Electric BWR-6 nuclear steam supply system with a rated power level of 2894 MWt and a Mark III pressure suppression containment system with a design pressure of 15 psig. Construction of Unit 1 is about 85% complete and Unit 2 is about 3% complete. Construction of Unit 2 has been deferred indefinitely, and the Applicant's motion to sever the Unit 2 proceedings from Unit 1 licensing proceedings has been granted. Consequently, both the Committee and the NRC Staff have limited this review to Unit 1.

The Committee's review included an evaluation of the management organization, the operational staff, and the training program. The Clinton Power Station is the Applicant's first nuclear station and staffing for plant startup and operation is not yet complete. The Applicant, however, has made considerable progress and has a well-established training program. The NRC Staff will continue to monitor the Applicant's progress and expects to complete its review before fuel loading.

The Applicant is currently restructuring the construction and operational quality assurance and quality control organization in response to NRC Staff concerns. The revised organization will be reviewed and audited by the NRC Staff. The Committee wishes to be kept informed on this matter.

The Mark III suppression pool dynamic loads have been identified as an Outstanding Issue in the NRC Staff's review. The NRC Staff has provided the Applicant with a proposal for the appropriate design basis loads, and it appears that the Clinton design will be able to accommodate these loads. The Committee will continue to discuss, on a generic basis, the Mark III suppression pool dynamic loads with the NRC Staff.

Hydrogen control systems for Mark III containments are being developed by the Mark III Owners Group. Efforts by this Owners Group are being directed toward the development of a hydrogen ignition system which makes use of distributed ignition sources. The NRC Staff has indicated that they will be able to meet with the Committee on this matter in the near future. The Committee expects to review this system on a generic basis. Acceptability of this system is a License Condition.

The Applicant, in response to NRC Staff requirements, has reevaluated certain safety-related systems of the Clinton design using the ground motion parameters that describe the site-specific spectra equivalent to a design basis earthquake of M_b equal to 5.8. The Applicant has reanalyzed what he believes to be the limiting structures and components using this new response spectrum and has concluded that all Seismic Category 1 structures will withstand the design basis earthquake. Work by the Applicant is continuing. The Committee believes that specific attention should be given to the seismic capability of the emergency AC power supplies, the DC power supplies, and small components such as actuators and instrument lines that are part of the decay heat removal system. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

In its Safety Evaluation Report dated February 1982, the NRC Staff has identified a number of Unresolved Safety Issues as being applicable to Clinton as well as a number of Outstanding Issues, Confirmatory Issues, and License Conditions. We believe that if due consideration is given to these matters and to our recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Clinton Power Station Unit 1 can be operated at power levels up to 2894 MWt without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References

1. Illinois Power Company, et al., "Final Safety Analysis Report, Clinton Power Station Units 1 and 2" with Amendments 1-12.
2. U.S. Nuclear Regulatory Commission, "Safety Analysis Report Related to the Operation of Clinton Power Station Unit 1," NUREG-0853, dated February 1982.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 11, 1981

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON ENRICO FERMI ATOMIC POWER PLANT UNIT NO. 2

Dear Dr. Palladino:

During its 256th meeting, August 6-8, 1981, the ACRS completed its review of the application of the Detroit Edison Company (Applicant) for a license to operate the Enrico Fermi Atomic Power Plant Unit No. 2 (Fermi-2). A Subcommittee meeting was held in Washington, DC, on July 24, 1981 to consider this project. A tour of the facility was made on July 15, 1981. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee reported on the construction permit application for this unit in its report dated March 9, 1971.

The Enrico Fermi plant is located in Frenchtown Township, Monroe County, Michigan. The nearest population center is the city of Monroe, Michigan about 5.5 miles west-southwest of the site.

Fermi-2 is equipped with a General Electric BWR-4 nuclear steam supply system with a rated power level of 3292 MWt and has a Mark I pressure suppression containment with a design pressure of 62 psig. The Applicant has performed a detailed evaluation of the containment's ability to withstand LOCA and relief valve hydrodynamic loads as required by the NRC for the Mark I Containment Program. As a result of this evaluation, extensive modifications were required and are underway. However, since the evaluation was performed prior to the issuance of the NRC report delineating the Staff's acceptance criteria (NUREG-0661 - Safety Evaluation Report, Mark I Containment Long-Term Program - Resolution of Generic Technical Activity A-7), the design has not yet been shown to be completely in conformance with this report. The Applicant has made a commitment to perform a plant unique analysis on the basis of the NUREG-0661 criteria and other requirements established by the Long-Term Program, including in-plant confirmatory tests to assess loads resulting from safety relief valve operation. The Applicant will submit this analysis to the Staff for audit review upon its completion. Subject to the results of this analysis, the NRC finds the Applicant's evaluation generally acceptable. This matter should be resolved in a manner satisfactory to the NRC Staff prior to full power operation. We wish to be kept informed.

We note that Detroit Edison has acted as its own architect-engineer for this project. The Applicant stated that this arrangement will result in a valuable carry-over of knowledge as people transfer from construction to plant operation activities. The NRC Staff has reviewed the Applicant's organization and management structure and has expressed some concern about the personnel transition. The Staff recommends that care be taken to assure that quality of construction and safety of operations are not compromised during the transition. We concur in this recommendation. To address a concern over a lack of commercial nuclear power plant operating experience, the NRC Staff is requiring that the control room staff be augmented with vendor personnel during startup. We recommend that the NRC assure that these personnel remain on site for a period of time which permits the necessary operating experience to be obtained by the Applicant's Staff.

The Applicant described the program and the philosophy for training of personnel. Training has a high priority and a training simulator has been ordered to aid in this effort. The simulator will be used for operator training and will also be used to train other plant personnel including managers and supervisors. It will also be used to test ATWS operating procedures. The NRC has reviewed the Applicant's ATWS procedures and finds them generally acceptable. The NRC should assure that the ATWS procedures and the associated simulator training are well coordinated.

The Applicant discussed provisions to address station blackout. In the event of a loss of all offsite AC power and loss of all onsite emergency diesel generators, the Applicant can call on a self-starting turbine-generator located onsite. While we recognize that this additional power source further lowers the probability of a station blackout, we recommend that the NRC Staff assure that procedures exist to address a station blackout event and that operating personnel are adequately trained in the use of these procedures. We wish to be kept informed.

Construction of this unit has taken a longer than usual time owing to financial difficulties and the impact of the TMI-2 accident. As a result, the Applicant has been required to perform a seismic reassessment of the structures, systems, and components required for safe shutdown based on currently accepted NRC design response spectra. This reassessment is still under way. Preliminary results indicate that there is sufficient margin in the original design to meet the NRC requirements and that only minor equipment changes will be required. This matter should be resolved to the satisfaction of the NRC Staff.

The NRC has begun review of the Applicant's emergency planning. Because of the plant's location, interaction with Canadian authorities is necessary. Responsibility for this interaction rests with the offices of the Federal Emergency Management Agency.

The NRC Staff proposes to require the installation of core thermocouples in Fermi-2 as specified by Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The Applicant has not yet agreed to this requirement. The ACRS supported use of core thermocouples in BWRs in its letter of November 10, 1980 to the NRC Executive Director for Operations, but called attention to the need for further study to determine the appropriate vertical location of such thermocouples. Since most of the information of interest from thermocouples may be obtainable from a small number of thermocouples placed in a more accessible location, we recommend that this requirement be reevaluated.

The Applicant's security plan was discussed. We note with approval that security guards will be Detroit Edison employees.

As part of the NRC Staff review of plant fire protection provisions, the Applicant simulated a control room fire to demonstrate that a fire external to the control panels will not result in a loss of redundant shutdown functions. The NRC Staff has identified what it believes to be deficiencies in the test and the Applicant has responded in a recent submittal. We believe this item should be resolved in a manner satisfactory to the NRC Staff.

Other issues have been identified as Outstanding Issues in the NRC Staff's Safety Evaluation Report dated July 1981. These include some TMI Action Plan requirements. We believe these issues can be resolved in a manner satisfactory to the NRC Staff and recommend that this be done.

The Committee believes that if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Enrico Fermi Atomic Power Plant Unit No. 2 can be operated at power levels up to 3292 MWt without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Detroit Edison Company, "Enrico Fermi Atomic Power Plant Unit 2 Final Safety Analysis Report," Volumes 1 - 11 and Amendments 1-37.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Enrico Fermi Atomic Power Plant Unit No. 2," USNRC Report, NUREG-0798, dated July 1981.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report, Mark I Containment Long-Term Program - Resolution of Generic Technical Activity A-7," USNRC Report, NUREG-0661, dated July 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 18, 1981

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON VIRGIL C. SUMMER NUCLEAR STATION UNIT 1

Dear Dr. Hendrie:

During its 251st meeting, March 12-14, 1981, the ACRS completed its review of the application of the South Carolina Electric and Gas Company for a license to operate the Virgil C. Summer Nuclear Station Unit 1. This project was considered at subcommittee meetings on February 26-27, 1981 in Columbia, South Carolina, and on March 11, 1981 in Washington, D.C. A tour of the facility was made by members of the Subcommittee on February 26, 1981. During its review the Committee had the benefit of discussions with representatives of the Applicant, the NRC Staff, the U.S. Geological Survey, and of the documents listed. The Committee reported on the construction permit application for this plant in a letter to AEC Chairman Schlesinger dated November 15, 1972.

The Summer plant is located in Fairfield County, South Carolina, about 26 miles northwest of Columbia, South Carolina. The nearest community with more than 1000 residents is Winnshore, about 15 miles to the northeast. The plant is adjacent to the Monticello reservoir, which provides cooling water for the main condenser, as well as the ultimate heat sink.

The Summer plant employs a Westinghouse, three-loop, pressurized water, nuclear steam supply system. The containment is a cylindrical, carbon-steel-lined, prestressed concrete structure having a design pressure of 57 psig.

At the construction permit review stage, some of the ACRS consultants were reluctant to accept the position of the Regulatory Staff and its consultants that the 1886 Charleston earthquake could be clearly localized in the Charleston area with regard to recurrence and recommended that a somewhat increased seismic design basis be employed. The ACRS supported the Regulatory Staff position favoring a safe shutdown earthquake (SSE) acceleration of 0.15g. However, in separate reports to the AEC dated May 13, 1971 and May 16, 1973, the ACRS urged initiation of a seismic research program intended to provide a better understanding of the likely causes of earthquakes near Charleston as well as several other areas in the eastern United States. Considerable research has since been undertaken in the Charleston area, and an improved understanding of the possible causes of earthquakes in the eastern United States has been developed. However, there still exists more than one theory with regard to the source of the 1886 Charleston earthquake.

Since the construction permit stage, a new issue has arisen with regard to the choice of seismic design basis; namely, the potential for a moderate earthquake at the site resulting from reservoir-induced seismicity. The Applicant has studied seismic activity in the vicinity of the Monticello reservoir since it was filled in 1977, and combined the results of those studies with information about the local geology and hydrology in arriving at the conclusion that a maximum near-field earthquake magnitude of 4.0 should be considered in evaluating plant safety. The NRC Staff and its consultants have concluded that a near-field magnitude of 4.5 should be used. However, one member of the NRC Staff disagrees with the majority Staff position, suggesting that the available information does not rule out a somewhat larger reservoir-induced earthquake, and that a near-field earthquake having a magnitude of 5.0 to 5.3 should be used for assessing seismic safety.

The ACRS consultants agree that there does not exist a very good basis for choosing a specific near-field event, and generally support the use of a near-field magnitude of about five for evaluation of the plant.

Because it is difficult to judge that the probability of significant exceedence of the original SSE is sufficiently small, the ACRS has requested, and the Applicant has provided, information that indicates there is sufficient margin in the original design to cope safely with accelerations considerably larger than the SSE of 0.15g, including those which might arise from a near-field, magnitude 5 earthquake.

The Applicant's results to date regarding seismic design margin are reassuring. The ACRS recommends that these studies by the Applicant be extended to include all systems and components whose function is important to the assurance of the continuing removal of shutdown heat. Such studies need not be completed prior to operation of the Summer plant.

The discussions relative to the seismic issues at the Summer Nuclear Power Station raise certain questions that we believe should be addressed. These questions, which largely pertain to emergency preparedness, include the ability of certain key systems to function after a major seismic event. Included among such systems are the emergency alarm features to alert the public to an accident in the plant, meteorological and field radiation monitoring networks, communications, and emergency evacuation routes.

As a result of the continuing microseismic activity induced by the reservoir, the Applicant has, at NRC request, agreed to continue seismic monitoring for at least the next two years. We recommend that the NRC Staff assure that the monitoring program is not halted prematurely.

In its review of the Applicant's organization and management, the NRC Staff has identified several areas requiring attention, including the size of the engineering organization and the adequacy of experience with nuclear power reactors within the company, including hands-on operating experience within the operating organization. The Applicant has taken steps to obtain the services of outside groups to provide additional technical capability for the short term while the needed in-house capability is developed. Care should be exercised that, as part of this effort, sufficient technical breadth and independence exists among the members of the Nuclear Safety Review Committee for the plant.

We have previously recommended that probabilistic safety analyses be performed for all plants in operation or under construction. We believe that this recommendation is applicable to this unit, but that such studies need not be performed prior to licensing of the plant.

During construction of the essential service water intake structure and pump house, settlement well beyond that predicted was experienced. While the settlement of the structures appears to have halted, the NRC Staff is still evaluating information addressing the stability of the subsurface materials and foundations of the intake structure and pumphouse. This matter should be resolved in a manner satisfactory to the NRC Staff.

The ACRS believes that, if due consideration is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Virgil C. Summer Nuclear Station Unit 1 can be operated at power levels up to 2775 MWt without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. South Carolina Electric and Gas Company, "Final Safety Analysis Report, Virgil C. Summer Nuclear Station," Volumes I-XX and Amendments 1-22
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 1," USNRC Report NUREG-0717, dated February, 1981
3. Letter from J. Devine, USGS, to R. Jackson, NRC, in response to an NRC request for update on USGS information concerning occurrence of earthquakes similar to the 1886 Charleston event, dated December 30, 1980
4. Memorandum from A. Murphy, Site Safety Research Branch, NRC, to R. Jackson, Chief, Geosciences Branch, NRC, Subject: Recommendation of Maximum Reservoir-Induced Earthquake at the V. C. Summer Nuclear Station, dated February 6, 1980
5. "Comments from the Palmetto Alliance, Inc., by Michael Lowe on V. C. Summer Operating License Application Review by the NRC Advisory Committee on Reactor Safeguards," dated February 26, 1981
6. "Testimony Before the Advisory Committee on Reactor Safeguards Related to the Virgil C. Summer Nuclear Station," Ms. Ruth Thomas, received February 26, 1981

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

letters

December 11, 1979

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM LOW POWER OPERATION OF SEQUOYAH NUCLEAR POWER PLANT, UNIT 1

Dear Dr. Ahearne:

During its 236th meeting, December 6-8, 1979, the Committee considered a proposal for interim, low power operation of the Sequoyah Nuclear Power Plant, Unit 1. At its 229th meeting, May 10-12, 1979 and also at its 228th meeting, April 5-7, 1979 the Committee had considered aspects of the application of the Tennessee Valley Authority (hereinafter referred to as the Applicant) for authorization to operate the Sequoyah Nuclear Power Plant, Units 1 and 2. A tour of the facility was made by members of the Subcommittee on January 24, 1976 and the application was considered at Subcommittee meetings on March 12, 1979 and on November 5, 1979. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for this plant on February 11, 1970.

The Sequoyah Nuclear Power Plant is located on the west bank of the Tennessee River in Hamilton County in southeastern Tennessee approximately 17 miles northeast of the center of Chattanooga, Tennessee. Construction on Unit 1 is essentially complete and construction of Unit 2 is about 90% complete. Each unit will utilize a four-loop pressurized water reactor nuclear steam supply system having a power level of 3411 MWt and an ice condenser system enclosed within a free-standing steel containment vessel which is surrounded by a reinforced concrete shield building. The ice condenser system is similar to that used in the McGuire Nuclear Station and the Donald C. Cook Nuclear Plant. The Applicant has modified the ice condenser system as a result of the operating experience gained in the Donald C. Cook Nuclear Plant. The Applicant and the NRC Staff have made plans to monitor the performance of the ice condenser containments at the Sequoyah Nuclear Power Plant (Generic Item 63 in the ACRS report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979). The Committee recommends that such plans be implemented.

The Sequoyah Nuclear Plant will utilize 17x17 fuel assemblies. A surveillance program has been developed by the NRC Staff to follow the behavior of these assemblies, and data are being obtained from several plants now in operation in which such assemblies have been installed for test. Experience to date has been satisfactory. The Committee wishes to be kept informed of the results of the various 17x17 assembly inspections and test programs now under way.

The Sequoyah site is considered by the NRC Staff to be within the Southern Valley and Ridge tectonic province. The maximum historic earthquake within this tectonic province is the 1897 Modified Mercalli Intensity (MMI) VIII earthquake in Giles County, Virginia. During the construction permit review, the NRC Staff concluded that a modified Housner response spectrum anchored at 0.18g was acceptable as the safe shutdown earthquake. Since that time, the NRC Staff has adopted methods which would characterize an MMI VIII earthquake with the more conservative response spectrum specified in Regulatory Guide 1.60 anchored at 0.25g.

The Applicant, in response to NRC Staff recommendations, has evaluated the Sequoyah design using a site-specific safe shutdown response spectrum developed from North American and Italian strong motion records of appropriate magnitude and epicentral distance and has compared the probability of the safe shutdown earthquake being exceeded at Sequoyah to that at other Tennessee Valley Authority plants that meet the Standard Review Plan. It has been concluded that the risk of exceeding the present design spectrum and the risk of exceeding the site-specific spectrum are comparable and that the probability of exceeding the safe shutdown earthquake is not appreciably different from that for other plants in this region. The NRC Staff has reviewed the Applicant's evaluation and has concluded that the Sequoyah plant is adequate to withstand the effects of the safe shutdown earthquake without loss of its capability to perform required safety functions. The NRC Staff, to verify their judgments regarding structural and component design margins, has performed an audit of the design margins in representative critical sections of the reactor and auxiliary building structures and in representative components required for safe shutdown.

The Committee recommends that this program for the quantification of the seismic design margin be continued and expanded to the extent necessary to ensure that all structures and equipment necessary to accomplish safe shutdown do indeed have some margin. Similar recommendations have been made by the Committee for the North Anna Power Station, Units 1 and 2, and the Davis-Besse Unit 1 in its reports dated January 17, 1977 and January 14, 1979. This matter should be resolved on a schedule and in a manner satisfactory to the Staff.

The Emergency Core Cooling Systems (ECCS) for the Sequoyah Nuclear Plant incorporate the Upper Head Injection (UHI) system. The NRC Staff has completed its review of the Westinghouse Electric Corporation ECCS evaluation model for plants equipped with UHI, and the Committee in its April 12, 1978 report on the McGuire Nuclear Station has concurred with the

Staff's conclusions. The NRC Staff has completed its review of the application of this approved evaluation model to the Sequoyah Nuclear Plant and concurs with the Applicant.

The Committee has been reviewing the circumstances relating to the recent accident at the Three Mile Island Nuclear Station Unit 2 and has made recommendations for improvements in plant design and operating procedures which should be considered for all pressurized water reactors. The Committee is continuing its review of the implications of this accident and expects to provide additional recommendations. It is expected that these recommendations will be considered and implemented as appropriate by the NRC Staff. The Committee wishes to be kept informed.

The NRC Staff has identified a number of outstanding issues, confirmatory issues, and licensing conditions, not related to TMI-2 accident considerations, which have not been specifically addressed in this report. These issues should be resolved in a manner satisfactory to the NRC Staff.

Various generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979. Those problems relevant to the Sequoyah Nuclear Plant should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: 54-60, 63-65, 69, 71, 72, 74, and 76.

The NRC Staff has not completed its review of the Sequoyah Nuclear Power Plant application for a normal operating license at full power, and various implications of the Three Mile Island accident on the Sequoyah Plant remain to be decided. The ACRS has not completed its own review in regard to these matters.

The Applicant has proposed a program of interim low power operation to provide improved operator training and the development of additional experimental information on the behavior of a nuclear unit and its systems under transient conditions. The Applicant has proposed a special test series which includes the following:

1. Natural circulation following a simulated reactor trip.
2. Natural circulation following a simulated loss of offsite power.
3. Natural circulation with loss of pressurizer heaters.
4. Effect of steam generator isolation on natural circulation.
5. Natural circulation at reduced pressure.
6. Cooldown capability of the charging and letdown system.

7. Heat removal following a simulated loss of onsite and offsite AC power.

8. Establishment of natural circulation from stagnant flow conditions.

9. Boron mixing and cooldown.

The NRC Staff plans to review the proposed experimental program in detail to assure itself that all safety-related aspects are being dealt with appropriately. The Committee wishes to be kept informed.

The NRC Staff advised the Committee that it will require that TVA's emergency procedures for Sequoyah be reviewed by Westinghouse. The NRC Staff also stated that an acceptable emergency plan will exist prior to reactor operation.

The Committee believes that there is reasonable assurance that the Sequoyah Nuclear Power Plant, Unit 1 can be operated on an interim basis up to power levels of about five percent of full power without undue risk to the health and safety of the public. Subject to approval of the detailed test program by the NRC Staff, the Committee recommends approval of an interim low power license for the purposes proposed.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

References:

1. Tennessee Valley Authority, "Final Safety Analysis Report, Sequoyah Nuclear Power Plant," Volumes 1 to 13, and Amendments 1 to 61.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the operation of Sequoyah Nuclear Plant Units 1 and 2," NUREG-0011, March 1979.
3. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated October 31, 1979, containing revised responses to the Lessons Learned Requirements.
4. Letter, L. M. Mills, TVA, to L. S. Rubinstein, NRC, dated October 30, 1979, containing responses to ACRS questions.
5. Letter from L. M. Mills, TVA, to L. S. Rubinstein, NRC, dated October 23, 1979, containing information on natural circulation in Sequoyah, Unit 1, and Diablo Canyon, Unit 1.
6. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated October 12, 1979, containing responses to ACRS recommendations.

7. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated September 7, 1979, containing responses to the Short-Term Recommendations of the Lessons Learned Task Force.
8. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated July 12, 1979, containing responses to NRC-I&E Bulletin 79-06A and ACRS recommendations.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 14, 1977

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

Dear Mr. Rowden:

At its 201st meeting, January 6-8, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application by the Toledo Edison Company and the Cleveland Electric Illuminating Company for a license to operate the Davis-Besse Nuclear Power Station, Unit 1. Members of the Committee visited the plant on May 18, 1976, and a subcommittee meeting was held in Washington, D.C. on December 21, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Babcock and Wilcox Company, the Bechtel Corporation, and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for this unit on August 20, 1970.

The Davis-Besse Nuclear Power Station, Unit 1, is located on the southwestern shore of Lake Erie about midway between the cities of Toledo and Sandusky, Ohio. The minimum exclusion distance is 2400 ft. The low population zone, with a radius of two miles, included about 870 people in the 1970 census. The nearest population centers are Toledo (1970 population 383,818) and Sandusky (1970 population 32,674), both about 20 miles from the plant.

The nuclear steam supply system employs a Babcock and Wilcox pressurized water reactor similar in most respects to those first used in the Oconee Nuclear Station. This system differs from the Oconee units and several other similar units in that the steam generator loops are raised about 30 ft above the level in the original plant arrangement. Although this change was made to eliminate the need for internal vent valves, four such valves are provided because of their beneficial effect in reducing steam binding following a postulated loss-of-coolant accident.

The proposed power level for the unit is 2772 MWt, as compared to 2633 MWt proposed at the construction permit stage. This higher power level is the same as that proposed for the Rancho Seco and Three Mile Island, Unit 2 reactors, both of which have been reviewed by the NRC Staff and the Committee and found acceptable.

The structures and components of Davis-Besse, Unit 1, were designed for a Safe Shutdown Earthquake (SSE) acceleration of 0.15g at the foundation level. Because of changes in the regulatory approach to selection of seismic design bases, the Committee believes that an acceleration of 0.20g would be more appropriate for the SSE acceleration at a site such as this in the Central Stable Region. The Applicant presented the results of preliminary calculations concerning the safety margins of the plant for an SSE acceleration of 0.20g. The Committee recommends that the NRC Staff review this aspect of the design in detail and assure itself that significant margins exist in all systems required to accomplish safe shutdown of the reactor and continued shutdown heat removal, in the event of an SSE at this higher level. The Committee believes that such an evaluation need not delay the start of operation of Davis-Besse, Unit 1. The Committee wishes to be kept informed.

The performance of the Emergency Core Cooling System (ECCS) has been evaluated using a Babcock and Wilcox evaluation model applicable to the raised-loop configuration. The NRC Staff has reviewed these evaluations and has determined that certain assumptions regarding return to nucleate boiling do not comply strictly with the provisions of Appendix K to 10 CFR Part 50. The NRC Staff is also reviewing several other areas relating to ECCS performance. These matters should be resolved in a manner satisfactory to the NRC Staff.

In conjunction with the evaluation and assessment of the impact of routine waste releases from this plant, the Committee recommends that the NRC Staff provide leadership in encouraging the development of improved environmental radiation surveillance capabilities on the part of the State of Ohio and appropriate local regulatory agencies.

The Committee notes that post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptible to ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The Committee believes that appropriate test and maintenance procedures should be developed to assure continuous long-term seal capability.

The Committee recommends that, prior to commercial power operation of Davis-Besse, Unit 1, additional means for evaluating the cause and likely course of various accidents, including those of very low probability, should be in hand in order to provide improved bases for timely decisions concerning possible off-site emergency measures. The Committee wishes to be kept informed.

The question of whether the design of this plant must be modified in order to comply with the requirements of WASH-1270, "Technical Report on Anticipated Transients Without Scram (ATWS) for Water-Cooled Reactors," remains an outstanding issue pending the NRC Staff completion of its review of the Babcock and Wilcox generic analyses of ATWS. The Committee recommends that the NRC Staff, the Applicant, and the Babcock and Wilcox Company continue to strive for an early resolution of this matter in a manner acceptable to the NRC Staff. The Committee wishes to be kept informed.

Davis-Besse, Unit 1, has installed a bypass loop containing two manually operated valves around the decay heat removal system suction line isolation valves. The normally closed bypass valves would be opened in the event of a spurious closure of one of the decay heat removal system suction line isolation valves during system operation. The Committee recommends that further attention be given to the means employed for isolation of the low pressure residual heat removal system from the primary system while the latter is pressurized, and that reliable means be developed to assure such isolation. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

The Committee supports the NRC Staff program for evaluation of fire protection in accordance with Appendix A to Auxiliary and Power Conversion Systems Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." The Committee recommends that the NRC Staff give high priority to the completion of both owner and staff evaluations and to recommendations for Davis-Besse, Unit 1, and for other plants nearing completion of construction in order to maximize the opportunity for improving fire protection while areas are still accessible and changes are more feasible.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 17, 1977

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON NORTH ANNA POWER STATION, UNITS 1 AND 2

Dear Mr. Rowden:

At its 201st meeting, January 6-8, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application of the Virginia Electric and Power Company for a license to operate North Anna Power Station, Units 1 & 2. This project was also considered during a Subcommittee meeting held in Washington, D.C., on January 5, 1977. The Committee previously completed a partial review of this project at its 198th meeting, October 14-16, 1976, as discussed in its report to you, dated October 26, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Virginia Electric and Power Company, the Westinghouse Electric Corporation, the Stone and Webster Engineering Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

In its report of October 26, 1976, on North Anna, Units 1 & 2, the ACRS had not completed its review of the adequacy of seismic design bases and seismic design; loss-of-coolant accidents and emergency core cooling; quality assurance and control of on-site fabrication and installation; asymmetric loads on pressure vessel structures arising from certain postulated pipe breaks; and plans for upgrading protection against fires.

The NRC Staff has now completed its review of the Stafford fault zone and concluded that the available geological and seismological information supports the conclusion that the Stafford fault zone is not capable within the meaning of Appendix A to 10 CFR Part 100, and that the available information does not warrant any change in the previously approved seismic design bases for North Anna 1 and 2. Representatives of the U.S. Geological Survey concurred that there exists no definitive information showing significant movement during the last million years and that the fault is not capable. Consultants to the ACRS concur with this interpretation. While they generally find the current design bases acceptable for the already constructed North Anna plants, they have recommended that, in view of the uncertainties of knowledge concerning the sources of earthquakes in the Eastern United States, a minimum safe shutdown earthquake (SSE) of 0.2g acceleration should be utilized for new plants for which construction permit applications are submitted in the future.

The Applicant presented partial information concerning the calculated safety factors during safe shutdown earthquake conditions for some of the engineered safety features. The Committee recommends that the NRC Staff review this aspect of the design in detail and assure itself that significant margins exist in all systems required to accomplish safe shutdown of the reactors and continued shutdown heat removal, given an SSE. The Committee believes that such an evaluation need not delay the start of operation of North Anna 1 and 2. The Committee wishes to be kept informed.

The NRC Staff has now completed its review of emergency core cooling system performance and found it to be acceptable. The Committee concurs.

The NRC Staff has conducted and is continuing extensive investigation of construction activities of North Anna Units 1 and 2. These investigations have been separated into four phases:

1. investigation of specific allegations made by three individuals of faulty construction practices;

2. a detailed inspection of certain safety-related piping not directly implicated in the original allegations but which was potentially subject to similar problems;

3. detailed monitoring of the nondestructive preservice baseline examination of selected welds in safety-related piping by the Licensee and his contractors; and

4. inspections of the performance of selected components in specific piping systems during the preoperational testing program.

The NRC Staff has concluded that various items of non-compliance with NRC requirements have occurred and has defined a program to remedy the matter.

The Committee has had the benefit of a review and evaluation of this matter by its own consultant, who supports the adequacy of the NRC investigations and has made several recommendations, including one related to a program to ascertain that significant deficiencies do not exist in safety related piping systems. The ACRS concurs. The Committee wishes to be kept informed regarding resolution of these recommendations.

The NRC Staff has reported that the matter of asymmetric loads on pressure vessel structures is essentially resolved. The ACRS has had the benefit of meetings of an Ad Hoc Working Group on this general subject, in Toronto on August 5, 1976, and in Los Angeles on December 1, 1976. The Committee agrees that, subject to final evaluation by the NRC Staff, this matter is in an acceptable status for North Anna 1 and 2.

The Applicant is in the process of studying fire protection measures at the plant in accordance with the guidelines of Appendix A to Auxiliary and Power Conversion Systems Branch Technical Position 9.5-1. The NRC Staff has stated that, as a plant about to come into operation, North Anna 1 and 2 will be given priority in the evaluation of fire protection matters, and that most, if not all improvements will be implemented prior to the start of operation on the second fuel cycle. The Committee finds this approach to be acceptable.

The Committee notes that post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptible to ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The Committee believes that appropriate test and maintenance procedures to assure continuous long-term seal capability should be developed.

The ACRS believes that, if due regard is given to the items mentioned above and in its report of October 26, 1976, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the North Anna Power Station, Units 1 and 2, can be operated at power levels up to 2775 MWt without undue risk to the health and safety of the public.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachment:
Report of W.R. Gall, ACRS Consultant, dated January 3, 1977, Subject: Review of Allegations and Inspectors Findings as Reported in NRC Investigation Report #50-338/76-28, 50-339/76-16 North Anna, Units 1 and 2.

REFERENCES:

1. North Anna Power Station, Units 1 & 2 Final Safety Analysis Report, with Amendments 1 through 60.
2. Safety Evaluation Report (NUREG-0053) related to operation of North Anna Power Station, Units 1 and 2, with Supplements 1 through 5.
3. Virginia Electric and Power Company (VEPCO) letter Serial No. 338 to Mr. Benard C. Rusche, ONRR, NRC, dated November 24, 1976, on environmental testing of safety related instrumentation.
4. VEPCO letter Serial No. 350 to Mr. Benard C. Rusche, ONRR, NRC, dated November 30, 1976, forwarding a document entitled, "Safety Related Equipment Temperature Transients During the Limiting Main Steam Line Break."
5. VEPCO letter Serial No. 346 to Mr. Benard C. Rusche, ONRR, NRC, dated November 30, 1976, on measures considered for use at North Anna re overpressurization events.
6. VEPCO letter Serial No. 316A, dated December 3, 1976, re model testing of LHSI pumps.
7. VEPCO letter Serial No. 298/102276, dated December 16, 1976, containing information on LOCA effects on reactor fuel. (Westinghouse PROPRIETARY).
8. NRC letter of December 14, 1976, from D.B. Vassallo to Dr. Dade W. Moeller, Chairman, ACRS, subject "Staff Report - Assessment of the Stafford Fault Zone."
9. NRC memo dated December 2, 1976, from Dudley Thompson and Boyce H. Grier to Ernst Volgenau, I&E, subject, "Transmittal and Evaluation of Investigation Report, No. 50-338/76-28, 50-339/76-16 - North Anna Units 1 and 2."
10. VEPCO letter Serial No. 371, dated December 9, 1976, forwarding a copy of VEPCO's reply to E. Volgenau re I&E Investigation Report Number 50-338/76-28 and 50-339/76-16.
11. NRC letter dated December 6, 1976 from E. Volgenau , I&E, to VEPCO Attn: Mr. T. Justin Moore, President referring to the I&E investigation of construction activities at North Anna 1 and 2 forwarding a "Notice of Violation", and a "Notice of Proposed Imposition of Civil Penalities."

REFERENCES (con't)

12. USNRC, IE Investigation Report 50-338/76-28, 50-339/76-16, Subject: "Investigation of alleged discrepancies in the construction and quality control program for piping installation at the North Anna Power Station."
13. VEPCO letter serial 390 to Dr. Dade W. Moeller, Chairman, ACRS, forwarding a copy of Mr. T. Justin Moore's letter of December 23, 1976 to Dr. Ernst Volgenau re the North Anna investigation.
14. VEPCO letter Serial No. 391, dated January 4, 1977, providing information re concerns related to auxiliary power and containment systems.
15. North Anna Environmental Coalition (NAEC) letter dated January 5, 1977, to Dr. Dade W. Moeller and Dr. David Okrent, ACRS, requesting that certain items be made a part of the record of the January 6-8, 1977, ACRS meeting.
16. NAEC letter dated January 7, 1977, to Dr. Dade W. Moeller and Dr. David Okrent, ACRS, adding two additional items to the list submitted in the NAEC letter of January 5, 1977.