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NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF:

THREE MILE ISLAND
SPECIAL INQUIRY INTERVIEW

INTERVIEW OF JOHN ANGELO

POOR ORIGINAL

Place - Bethesda, Maryland

Date - Monday, September 17, 1979 Pages 1 - 90

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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Interview of:
JOHN ANGELO
- -x

<u>NRC/TMI SPECIAL INQUIRY</u>

Room P-500
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland

Monday, September 17, 1979

The interview commenced at 9:30 a.m., pursuant to notice.

Present: John Angelo, William Parler, and Tom Cox.

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C O N T E N T S

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P R O C E E D I N G S

Whereupon,

JOHN ANGELO

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

EXAMINATION

BY MR. PARLER:

Q Please state your full name for the record.

A My name is John Angelo.

Q At this point I just want to ask you, Mr. Angelo, if you have received a letter from Mr. Rogovin, the director of the Special Inquiry Group. Prior to the commencement of this deposition I asked you that question. I don't believe that you did receive such a letter; is that correct, sir?

A No, I did not receive a letter.

Q I checked to the extent that I could my records, and it's not entirely clear to me that a letter was sent. The letter should have been sent about two weeks prior to this date. On September the 5th, because of the hurricane that came through, and the condition of the roof at 6935 Arlington Road, we were -- that is the Special Inquiry Group -- were evicted from our offices, and I have no way of checking and making certain that the letter was sent.

In any event, you haven't received such a letter. I did show you a copy of the letter to read, and I gather you've done so; right?

A Yes, I have read the letter.

Q Since I don't have a copy of the letter addressed to you, I cannot -- which is normally the case at this point in the record -- mark the letter as an exhibit.

I will for the record read the content of such a letter. The letter says, in pertinent part:

"The deposition will be conducted by members of the NRC Special Inquiry Group on Three Mile Island."

Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

BY MR. PARLER:

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

"Unless you have been served with a subpoena, which you have not," Mr. Angelo, "your participation in this deposition is voluntary, and there will be no effect on you if you decline to answer some or all of the questions asked you.

"However, the Special Inquiry has been given the power to subpoena witnesses to appear and testify under oath or to prepare and product documents, or both, at any designated place.

"Any person deposed may have an attorney present, or any other person he wishes accompany him at the deposition as his representative.

"The Office of the General Counsel of the NRC has advised us that it is willing to send an NRC attorney to all depositions of NRC employees who will represent you as an individual rather than represent NRC.

"Since the NRC attorney may attend only at your affirmative request, you should notify Richard Mallory, 634-3224, in the Office of the General Counsel as soon as practicable, if you wish to have an NRC attorney present.

"You should realize that while we will try to respect any request for confidentiality in connection with publication of our report, we, that is the Special Inquiry Group, can make no guarantees.

"Names of witnesses and the information they provide may eventually become public inasmuch as the entire record of this Special Inquiry Group's investigation will be made available to the NRC for whatever uses it may deem appropriate. In time this information may be made available to the public voluntarily or become available to the public through the Freedom of Information Act.

"Moreover, other departments and agencies of the government may request access to this information pursuant to the Privacy Act of 1974.

"The information may be also made available in whole or in part to committees or subcommittees of the U.S. Congress."

The letter also provides that if you have testified previously with respect to the Three Mile Island accident, it will be useful if you could review any transcripts of your previous statements prior to the deposition.

"Thank you for your cooperation. Sincerely, Mitchell Rogovin, Director, NRC/TMI Special Inquiry Group."

Now, as I have already said, you did not receive this letter, as you were supposed to receive, apparently because it was not dispatched by the Special Inquiry Group.

I read the letter to you. Is it agreeable to you to proceed with this deposition?

A Yes.

Q Do you understand the information set forth in the letter, including the general nature of the NRC/TMI Special

Inquiry, your right to have an attorney present here today as your representative, and the fact that the information you provide here may eventually become public?

A Yes.

Q Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

BY MR. PARLER:

Q Mr. Angelo, is counsel representing you presently today?

A No.

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

Mr. Angelo, if at any time during the course of this interview, you feel you would like to be represented by counsel and have counsel present, please advise me, and we will adjourn these proceedings to afford you the opportunity to make the necessary arrangements and, of course, the necessary arrangements are as provided in the letter, calling Richard Mallory.

Is this procedure agreeable to you?

A Yes.

Q I would note at this point that Mr. Cox, a member of the Technical Staff, the Special Inquiry Group, has joined us for this deposition. It's Mr. Tom Cox.

Mr. Angelo, you should be aware that the testimony that you give has the same force and effect as if you were testifying in a court of law. My questions and your responses are being taken down, and they will later be transcribed. You will be given the opportunity to look at that transcript and make changes that you deem necessary.

However, to the extent that your subsequent changes are significant, those changes may be viewed as affecting your credibility, so please be complete and accurate as you can in responding to my questions now.

If you at any point during the deposition don't understand the question, please feel free to stop and indeed stop me and indicate that, and we will make the necessary clarification at that time before we proceed on the record.

Let me inform you of two basic ground rules:

One is that you permit me to finish my questions before you give your response, even if you know what the question is going to be, because the reporter cannot take down both of us speaking at the same time.

Second, please respond verbally and audibly. Motions such as nodding your head cannot be taken down by the reporter.

Now I understand, since you did not get the Special Inquiry Group's letter, you did not bring a copy of your resume to the deposition.

However, you did prepare a copy of your resume for participation in an Atomic Safety & Licensing Board hearing which I did bring and which I now show to you.

This is a two-page document with the witness' full name at the top of the first page.

I will ask you if this document, which I have just given o you, accurately summarizes your educational and employment background, and if it needs to be updated because of events which have happened since the time that this statement of qualifications was prepared, please update it.

A Yes, the document is up to date and accurate.

Q All right. Off the record.

[Discussion off the record.]

BY MR. PARLER:

Q Mr. Angelo, I will mark your statement of qualifications for identification as Exhibit 1065.

[The document referred to was marked Exhibit 1065 for identification.]

BY MR. PARLER:

Q Have you made any prior statements or been asked to give a statement in connection with events that have happened after the Three Mile Island accident on March 28th, '79?

A No. Do you mean statements to the --

Q To the President's Commission, for example, or to an official governmental body, or to the ACRS.

A No.

Q What is your current position, Mr. Angelo, in the NRC?

A I hold the position of Senior Licensing Project Manager in the Division of Project Management, and have been assigned as Task Manager for Generic Task No. A-17, which is titled "Systems Interaction in Nuclear Power Plants."

Q Off the record.

[Discussion off the record.]

BY MR. PARLER:

Q That was also the position that you had several months ago, say around March the 30th, 1979?

A Yes, the same position.

Q What is your educational background, that is your degree in your major field? I realize that's in your statement of professional qualifications, but for the record, at this point, would you so indicate?

A I have a Bachelor of Science in Electrical Engineering from the University of Idaho in 1949, and I hold a Master's Degree in Engineering from Union College in 1963.

Q Andyour employment background or your position with the NRC has been the position that you have described, Senior Project Manager in the Division of Project Management, that is from January 1975, when the NRC was created?

A Yes, it has been the same position.

Q And prior to that, you worked with the Atomic Energy Commission?

A Yes, and in the same position.

Q All right.

Incidentally, within the Division of Project Management, who is your supervisor? In other words, what is your chain,organizational chain? That is as of, say, March '79?

A Okay, my immediate supervisor is John F. Stolz, Branch Chief of Light Water Branch No. 1.

Q And Mr. Stolz reports to an Assistant Director?

A Yes. Mr. Stolz in turn reports to, at the present time it is Steve Varga, but in March it was Dominic Vasallo.

Q Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

BY MR. PARLER:

Q I believe that you have already indicated, Mr. Angelo, that one of your duties is to serve as task manager for Task A-17, which, as I understand it, has to do with a study of systems interaction in nuclear power plants. Is that correct, sir?

A Yes, that is correct.

Q You were assigned as task manager for that project approximately when?

A In June 1977.

Q At the outset, I think it would be helpful if the record would indicate at this point what the words "systems interaction" mean in the context we are talking about; that is the regulatory review and licensing of a commercial nuclear power plant.

In other words, what is systems interaction?

A I could best define systems interaction as an event that may occur in one system that has an adverse effect on the performance of other systems.

By adverse effect, we mean some effect that would seriously or substantially degrade the safety performance of the other systems.

Q And the systems that are involved need not be necessarily limited to those systems that are a part of the nuclear steam supply system; is that correct? It could be the systems that are a part of the total nuclear power plant?

A Yes. It could be any system in the nuclear power plant that has been determined in the course of this study to perform a vital safety function.

Q Well, then, since the term "systems interaction" covers the entire plant, it would seem that the study is a very broad one. That is the study of systems interaction -- interactions, and at this point I would like to ask you, are you concerned in your study with all possible systems interactions, or is the study bounded in some respects?

A No, because of its very nature, we spend a considerable amount of time to bound the study, so that it could be practically achieved, or so that it could yield some practical results. So we deliberately bounded the study rather severely.

Q Go ahead. I was going to ask you -- go ahead and state to the best of your recollection how that was done, if you don't mind, sir.

A The way we bounded the study was to give consideration to the comments and letters made by the Advisory Committee on Reactor Safeguards, and we attempted to draw from that, from those letters and comments, some understanding of what their concern was, and then we proceeded to apply our judgment as to what we thought we could practically achieve, so in the course of four or five months, I believe, we attempted to define a scope of work that was generally agreed to and found reasonable by most of the division directors of NRR.

We eliminated from the scope of work such broad categories of interactions as operator errors, design errors, maintenance and installation errors.

We recognized that these could, in a very broad sense, be termed system interactions, but we were concerned more with trying to develop a method of handling the broad field of interactions and to demonstrate quickly that we could develop such a method. So we limited the scope of work deliberately then to non-accident conditions in the power plant.

That is, we limited it to the kinds of things that could be expected to occur on a, more like a day-to-day basis, because we interpreted this to be the concern of the ACRS.

So, in addition to eliminating the other things from our scope of work, we also eliminated fires, earthquake, flood, tornadoes, and accidents such as pipe ruptures.

Q Well, with the things that you have eliminated, those are clear now, you have also mentioned that you were concerned with non-accident conditions.

Could you, for the record at this point, indicate the kinds of things that the study is concerned with?

A Yes. The kinds of things the study is concerned with is things that are called normal transients and transients of rather frequent occurrence, such as loss of offsite power, for an example, trip-out of the generator set, normal start-up and shutdown of the plant, where safety systems are called on to remove things like core decay heat, control the reactor criticality, and maintain the integrity of the reactor coolant pressure boundary.

These sorts of things are an occurrence on a day-to-day basis in a power plant. We thought that these were the most fruitful areas to pursue, mainly because our feeling was that the accident conditions had been very thoroughly reviewed in comparison, that is, to non-accident conditions, although this isn't to infer that we believe non-accident conditions were slighted.

It is simply to emphasize that we thought much more attention had been given to the accident conditions than to the non-accident, day-to-day conditions.

We are particularly concerned that any interactions that occur or were possible on a day-to-day basis did not progress into an accident condition, so the main thrust of our work on system interaction was and is being directed to those conditions which have the potential of propagating to a more serious condition.

Q Is my understanding correct of one of your earlier responses that all operator actions and maintenance errors are excluded from the study?

A Well, they are excluded in the sense that we didn't deliberately go out and look for operator errors or maintenance errors, although a large number of them are accounted for in the system interaction in this sense, that if we look at, let's say, the failure of a valve to operate, it could be just as well interpreted that the operator has caused the failure as it could that a mechanical or electrical system has caused the failure.

But we didn't go out and deliberately insert into the program things that -- things that would be only postulated to be the result of an operator action.

Q The word "system," Mr. Angelo, may mean different things to different people, and in the interest of having a record which is as clear and unambiguous as it can be in a complicated area, could you define or indicate with some precision what the word "system" means for purposes of this study that you were talking about?

Maybe you've already done that, but maybe you could shed a little bit more light on that.

A Well, I've never come across a definition of a system that would be accepted by most of my peers, but I believe I can make a definition that makes sense and that is that a system is a collection of components that function together in such a way as to perform a well-defined function.

I could give you an example. For example, let's take the system that seems to be on lots of people's minds these days, and that is the auxiliary feedwater system. That's a system that consists of pumps, valves, and a supply of water. Its function is to deliver water to a steam generator or a group of steam generators in the absence of normal feedwater system performing that function.

As it turns out, in our study of system

interactions, the definition of the system and its boundaries seems to be less and less important as we go along. Initially when we started it, started our project, we thought that it would be very important to define the system, its function, and its boundaries. But as we go along in this project, we find that that's less and less important.

What seems to be more important now are the components of systems that do the job, and that is because components appear in various combinations to perform more than one function, and essentially appear in more than one system.

So I would have to guess that a definition of a system is not really very important. What is important now is a definition of functions and identification of components that perform those functions.

Q How are redundant systems being treated for purposes of the systems interactions study?

A Redundant systems are really treated as two separate and independent systems in that it is important to treat them that way because we are particularly interested in interactions from one of the redundant subsystems to the other redundant subsystem.

One of our principal criteria for safety is redundancy in systems, and that redundancy must be preserved, so the thrust of our work in system interaction is particularly directed toward interactions that occur among, you might say, subsets of redundant systems.

Q Could you summarize how the work on systems interaction is being conducted?

I should also state that you have some material with you, and feel free to consult that material at any time or read from it, if necessary, because this project has been going on for some time, and I would imagine that the details are rather voluminous.

So my question at this point was how, without covering all of the details but the significant points, is the work being conducted?

I would assume that within the NRC that there is some division of responsibility, but you are the task manager, as has been indicated, I would assume that some work is being done by contractor. I've heard the word Sandia Laboratories mentioned. I've seen some references to work at the Oak Ridge National Laboratory, I've seen some references to a Zion Plant interaction study, and I suppose I've also heard something about a systems interaction study in connection with the Indian Point 3 plant.

Now I go through those things to suggest to you some of the kinds of things that you might want to comment on, indicating for the record at this point how this project is being conducted.

A Basically the work is being done by -- under contract by Sandia Laboratories in Albuquerque, New Mexico. That work is technically monitored and directed by a group of persons within the NRC, principally from the Office of Nuclear Reactor Regulation and the Office of Standards Development, with assistance and consultant advice from the Office of Nuclear Reactor Research.

Sandia has been under contract to the NRC since May of 1978. Some of the other elements you mentioned are ancillary to the real program of system interaction.

For example, the Zion study was performed by Commonwealth Edison Company with assistance from Fluor, Pioneer -- I'm not sure of their name, I think it's Fluor Power Services now.

That study was very limited in its scope. That study was performed mostly at the request of the ACRS, I believe, and it concerned itself with a study of events that have occurred in nuclear power plants called Licensee Event Reports.

We in system interaction made use of some of the results of that study and Sandia also made use of the results of that study, in a sense that we used that study in order to make sure that our study would reflect the type of actual kinds of events that occurred in power plants.

We didn't want to model any particular chain of events, but we wanted to make sure that our study covered the kinds of things that happened on a more or less day-to-day basis in nuclear power plants.

The study at Oak Ridge was supposed to have been an essential adjunct to what we were doing at Sandia, but that study never got funded, and was never carried to completion. So whatever was attempted -- whatever we anticipated doing at Oak Ridge under a separate study, we are doing at Sandia now in a somewhat limited extent.

The Oak Ridge study was directed toward looking at specific interactions between two systems. That is, control systems and plant protection systems.

Our study will pick up the same kinds of interactions, but in a more general sense.

Q Excuse me for interrupting you, but what do you mean, "in a more general sense"? If you could elaborate on that.

A Well, that is we will not probe in as great a detail as we would have expected the Oak Ridge study to go. We probably won't go to all of the control elements in a nuclear power plant, but we will go far enough to identify either further work that might have to be done or at least to identify that we don't need to go any further in the control of certain components.

Q Who initiated the Oak Ridge National Laboratory study? Was that the Nuclear Regulatory Commission?

A Yes, Division of Operating Reactors is the group who initiated that study, but as I said, it never got funded, and no work was done beyond some initial scoping.

The Indian Point study, I don't believe, has ever progressed to a point where any definitive work was done, and my understanding -- and I was not present at the latest ACRS meeting, so I can't speak from first-hand knowledge of that, but I believe Indian Point study will have a different emphasis than the Zion Station study, and we may not be able to derive any direct use of that in our system interaction.

Q Do you have any information or understanding as to what the different emphasis in Indian Point will be, in the Indian Point study?

A Well, I only have this from hearsay.

Q Right.

A And that is that the Indian Point study has been recommended to be directed towards the design efforts -- in other words, how system interactions might be introduced by the designer, by the design of the plant, rather than by operation of systems.

Q As you indicated some minutes ago, matters involving design error is not within the scope of the systems interaction study that you were managing; isn't that correct?

A Yes, that's correct, although I think it's important to bear in mind that if an error has been committed in the design, that would lead to an interaction, I am sure we would be able to identify it.

What I meant by design errors is that if a designer, for example, has undersized a pump, we would not probably be able to identify that kind of a design error. Cur study proceeds on the assumption that the designer has correctly sized things, like pumps and pipes and tanks, and switch gear, and unless the error is very obvious, we probably would not find it.

Q Who initiated the Indian Point study, do you know?

A I believe the ACRS are the ones who asked that a study be made on Indian Point.

Q Do you have anything else to add or that you could add about the method, the approach that Sandia Laboratories is using for their contributions as a contractor to the NRC system interaction study?

A Yes. Sandia has chosen tc use a method that is commonly referred to as a fault tree method. We selected Sandia because of their demonstrated capabilities in this area, and particularly their demonstrated capabilities in safeguards systems -- I mean industrial security matters, and their work and the follow-on tc the reactor safety study, commonly referred to as the Rasmussen Study.

When I say demonstrated capabilities, I mean by that that they had demonstrated to a number of us in the NRC that they did possess the kind of abilities that would be needed to complete our project.

We did recognize that Sandia Laboratory personnel may lack some familiarity with the nuclear power plant, and we recognized that we and the NRC would have to provide that kind of specialized assistance.

But to get back to the fault tree method of analysis, I could describe that as a method of depicting or illustrating the ways in which faults can occur in any collection of components. That is if one analyzes a system by postulating all of the components in a failed state, and as you search for all the ways that components can fail, you place them all in a faulted state, and then proceed to identify the unique combinations of failures that could cause the loss of a safety function.

Now these combinations become very numerous. In fact, they can number up in the millions of combinations. So the system also uses a method of -- of very quickly and accurately reducing these millions of combinations down to the ones that you are very vitally concerned with.

A method of doing that is a method called system -- I mean the -- let's say system equation -- SETS, equation transformation system. SETS equation transformation system. It's a computer code that is uniquely developed to analyze fault trees.

Once these unique combination of faults are identified, then they are subjected to a search for any interaction or any characteristic that could cause those failures to occur.

Give you an example: If you were particularly -- if a set or group of failures involved, let's say, two pumps, we would then probe for all the ways in which an event could cause the same two pumps to fail.

If you find such an event, that is a system interaction, and that would be the main thrust of our concern.

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Q So as I understand what you have just said, the method that is being followed by the Sandia Laboratories to carry out their contractual responsibility for the system interactions study is basically a fault tree analysis approach, along the lines of the approach taken in the Rasmussen Study. Is that -- perhaps it's overly simplified, but is that the substance of what you said?

A Yes, I think you could say that it generally is the same technique, although a lot of the Rasmussen study was more event trees than they were fault trees, but the technique is exactly the same.

Q And the Staff in the conduct of its review of an application, that is the NRC's Regulatory Staff, follows a different approach; is that correct?

A Yes, the Staff's method of review doesn't make use of the fault tree method.

I may have to qualify that. I'm not sure that there aren't some members of the Staff who might think fault tree, without actually writing it all down. It's hard now to define in matters of what go on in the mind of the reviewer, whether he isn't using fault tree. I'd have to say there's no evidence to me that he actually goes through this complicated and very involved manipulations that we do at Sandia Laboratories. But the logic may still be there, the type of thinking might be there.

Q Generally speaking, is it correct that the Staff in its review of an application with the possible qualification that you have just given, evaluates an application or gets a set of criteria that approaches what is generally referred to as a deterministic approach instead of a probabilistic approach, or words to that effect?

A Well, yes, the Staff does -- normally doesn't a probabilistic approach, although in certain areas they do, in the matter of site accidents, for example, our probabilities analysis is part of the review, but in general in the review of plant systems for their performance, a probabilistic approach is not used. A deterministic approach is used.

But we are still not even in system interaction with fault trees, we are not generally introducing probability either. We are still using a deterministic approach in the analysis of systems.

The difference is that we put it all together and depict it in a logical fashion, if you will. In other words, by our method, you can graphically trace through the system performance and the faults of that system.

Q Which approach, as I would understand it, that it would be very helpful to take, because of the kind of problem that you are trying to come to grips with, the large number of possible interactions.

In other words, there are no criteria, nowhere near having a criteria for determining whether a -- what systems interactions are or which are acceptable and which are not.

Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record -- off the record.

[Discussion off the record.]

MR. PARLER: On the record.

BY MR. COX:

Q John, with regard to your statement of a few minutes ago in describing how the SETS program evaluated or identified what could be a large number of failure combinations for system interactions, you mentioned that that program or Sandia using the program had a method of reducing the system interactions identified down to a few that would be looked at.

How is this done in a general way that I can understand? How is that reduction in number of combinations achieved?

A The reduction from millions of potential -- I won't use the word interactions because at the first run of the SETS code, no interactions are even considered. We merely look at all the possible combinations of failures that could produce an undesirable result, regardless of whether those combinations are caused by interactions or whatever their cause may be.

That is the first printout of the SETS code. Then you input back to the SETS code descriptive characteristics. For example, you ask the code to print out all of the combinations of failures that are linked by a characteristic, a particular characteristic, let's say, power, electric power.

The SETS code then would take from these millions of combinations and print out only the combinations that are linked by power.

You could ask it then to print out all of the combinations linked by lubrication, that have lubrication as a characteristic, and location, and for location we have selected three gross locations.

The containment is one, the auxiliary building another, and any other place in the power plant as the third location.

So, for example, you would ask the computer to print out all of the combination of faults or failures that could occur because the components are in the containment, for example. So out of the millions of combinations, only a few hundred are linked by the characteristic of being inside the containment.

Out of the possible millions of combinations, only perhaps a few dozen are linked by lubrication.

Q Failure of lubrication?

A Yeah, by having the characteristic of lubrication.

Okay, then you take one of these -- for example, let's say that the code has taken the millions of component failure combinations and printed out, oh, several dozen that depend on lubrication to function.

Then you examine these now to see whether they have a common lubrication system. Of course, if the power plant is designed properly, there would be no -- among safety components, you would not expect to find two vital components linked by the same lubrication system. So out of the millions of combinations, you are left to examine only a few thousand in different categories of characteristics like lubrication, cooling, power supply, actuation, circuitry.

Q Is there any intent to use probabilities c failure, or reliabilities of components?

A We may eventually have to apply some kind of probability at the end, when we are left with a few interactions that are still likely to occur. And then we may examine them as to their probability of occurring and their significance, really.

We expect that we will apply this measure at the end to a very, very limited number of interactions. We would expect that -- and results so far demonstrate that the millions quickly -- the millions of combinations quickly converge to only a handful, a dozen or so, that would have to be examined, and perhaps probability might be the way.

We haven't decided yet, until we get the entire list of things we have to look at. We will apply other measures, for example, you may look at the number of times a component shows up in a combination, a component.

You may look at the number of events that it might take to cause the interaction to occur.

For example, we have found so far in our studies that most interactions would only occur or are possible to occur only if more than one event occurs. It takes generally two to three events to cause an interaction to occur.

So we might look at the number -- the number of ways that a component can fail as some measure of whether it should be retained as significant.

For example, certain components only have one or two likely ways of failing, but other components may have four or five ways.

For example, a pump can lose its lubrication, it can lose its water supply, it can have its shaft failure. There are many, many ways for a component like a pump to fail. There are relatively few ways for a component like a heat exchanger to fail.

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BY MR. PARLER:

Q It's my understanding, Mr. Angelo, from what you've said previously that the systems interaction task was initiated by the Staff at the request of the Advisory Committee on Reactor Safeguards.

Is my understanding correct?

A Yes.

Q When was the ACRS request made and in what form was it made, if you recall, or if you have a document there, please, sir.

MR. PARLER: Off the record while he's looking.

[Discussion off the record.]

MR. PARLER: Back on the record.

THE WITNESS: The first indication I had of this was a letter dated June the 17th, 1977 from Chairman of the ACRS, Mr. Bender, to Mr. Gossick, Executive Director for Operations. And in this letter, he recommended that the NRC perform a study on system interactions, and he gave some examples of possible studies.

BY MR. PARLER:

Q Would that letter happen to be referenced in a document called NUREG 0410, or do you know?

A No, I don't know whether that's -- I'm not familiar with the content of --

Q Do you have an extra copy of that letter or not?

Off the record.

[Discussion off the record.]

MR. PARLER: On the record.

BY MR. PARLER:

Q You said there was another record, another letter, Mr. Angelo, in addition to the June 17, '77 letter from Mr. Bender to Mr. Gossick?

A Yeah, approximately 10 days after the June 17th letter, a letter dated June the 28th from Mr. Fraley, Executive Director of ACRS, to Mr. Case, transmitted all the correspondence on system interaction from the ACRS, and this -- this letter, this latest June 28th letter, makes reference to 15 other letters.

Q I have heard there was a reference as early as 1974, a memorandum from the ACRS on systems interaction that was perhaps raised in connection with the ACRS review of a licensing proceeding. Not that it's overly important, but for context. Do you have any indication there as to how many years ago it was when the ACRS first raised the question of systems interaction?

A The first letter, which is one of the referenced letters I talked about before, is a letter dated November 8, 1974.

Q Was that on the Quad Cities case, or do you know?

A No particular case was mentioned, but it was a letter to Manning Muntzing, who was Director of Regulation at the time, from Mr. Stratton, who was Chairman of ACRS at the time.

This is the first correspondence that is identified as related to system interaction, although it's my understanding that the term or the problem may have been talked about before then, but this at least is the first physical evidence we have that attempts to define the problem.

Q May we borrow your book with your letters to make copies of, so that we can mark them for identification, please? Unless you object.

A No, I think this is all in the record, and really these 15 letters form the background of how we attempted to develop our study.

Q You mean in the record, you mean in the public record already, is that what you meant?

A That's right. These are all in the public record.

Q Well, I think it would be helpful for our purposes to have them all together in our record of this deposition which, as far as I am aware, will be the only record made, which deals with the subject of systems interaction for the Special Inquiry Group.

Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

The documents that we have been talking about from the Advisory Committee on Reactor Safeguards to the Staff I will now mark for identification.

There is a document from R. F. Fraley, Executive Director of the ACRS, to E. G. Case, Acting Director, Office of Nuclear Reactor Regulation, dated June 28th, 1977. I will mark this document for identification as Exhibit 1066.

Exhibit 1066 has attached to it 15 attachments. The first attachment is a letter to L. V. Gossick from M. Bender, dated June 16th, 1977, subject, review of systems interaction.

Attachments 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, and 14, all are exchanges of correspondence either from or to the Advisory Committee on Reactor Safeguards concerning various subjects relating to systems interactions.

Attachment 15 is a letter to L. M. Muntzing from W. R. Stratton, dated November the 8th, 1974, on the subject of systems analysis of an engineered safety system.

All of these documents are a part of Exhibit 1066. And it's my understanding, Mr. Angelo, that these documents, these 15 letters that you referred to earlier, are all of the -- certainly of the major correspondence between the ACRS and the NRC, or its predecessor, the Atomic Energy Commission that you are aware of on the subject of systems interaction; is that right, sir?

THE WITNESS: Yes, that's correct.

[The documents referred to were marked Exhibit 1066 for identification.]

BY MR. PARLER:

Q It is my understanding that at the beginning when this subject was discussed, that there was some effort that had to be made to try to find out what the Advisory Committee on Reactor Safeguards had in mind in this area; is that correct?

A Yes. There were several efforts we made to try to get some better definition of this. I would say, though, that we pretty much proceeded on the basis of attempting to make our own definition, rather than to rely so much on the ACRS to make that definition for us.

We believed that we read their concerns in this collection of letters and were able to make a definition that would go a long way towards resolving their concern. I don't believe that we -- that we tried to extract from the ACRS an exact definition because we recognized the difficulty of making such a sweeping definition.

Q I have a letter dated April the 30th, 1974 which presumably came from the files of the Advisory Committee on Reactor Safeguards. I don't know who the letter was written by, because that is -- has been deleted from the copy of the letter that I have.

The copy of the letter that I have, all of the names have been deleted.

[Handing document to witness.]

This letter would appear to raise certain questions concerning systems engineering. I would like to mark this letter for identification as Exhibit 1067.

[The document referred to was marked Exhibit 1067 for identification.]

BY MR. PARLER:

Q Have you ever seen that letter before?

A No, I haven't seen this before, but the marking CT-373 appears to be a consultant's copy. That's generally the way these letters are marked as coming from consultants to the ACRS.

Q I have marked this letter as Exhibit 1067 for identification s' ply to add to the other 15 letters that were refer-ed to in Exhibit 1066.

A far as you are aware from your fast perusal, does it seem to add any insights other than revealed by the other correspondence that you are familiar with and that you provided us with a copy of earlier?

Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

THE WITNESS: Well, I think the letter addresses in general the same kind of concerns that have been addressed in all of the other letters that came after this date of April the 30th, 1974, and in a sense the main concern expressed in this letter really defines the things we are concerned with in system interaction.

That is the letter mentions, and so does our task scope mention, the fact that system interactions are believed or at least felt to be introduced into the design because the design has to be broken down into a lot of disciplines, mechanical, structural, electrical, chemical, and a lot of scientific disciplines, such as geology, seismology, meteorology, hydrology, and the concern was whether all these different groups really coordinate their work and are aware of what each is doing, so that the design comes out as a well-coordinated design able to function with all its many systems and thousands of components.

I think that's the basic problem, if you have a big job to do, you have to break it up into specialized areas to get the job done. Then you are concerned with whether all the specialists worked together properly.

We had again -- that is a legitimate concern. I think it is a concern of all engineering managers. It's my opinion that the NRC in formulating the standard review plan demonstrated quite well the fact that the job of review, for example, is many disciplines.

I think our effort in system interaction is the attempt to cut across all of these disciplines and try to bring in one place all of the significant and pertinent areas that might be a cause of concern.

We have attempted again to limit it to more of the physical arrangement of the plant and we have left out purposely the human element, you might say.

Q When you say bring it into one place, I gather that you mean that in addition to the various technical disciplines and specialists that are in the review branch now, that eventually perhaps there should be some place in the organization that looks at the entire plant in an overall perspective from the standpoint of significant systems interactions issues. Is that what you had in mind? You didn't mean in lieu of systems interactions, in lieu of the technical reviewers that we now have, did you?

A I'm not sure I understand you. Do you mean are we in system interaction proposing a group that would look across all the disciplines? Is that what you --

Q Well, it's my understanding that at some point when the study is over and that it's implemented, it will be implemented to achieve certain objectives.

Now perhaps it is premature to even raise this question, but in the testimony that you gave, before I asked my last question, my recollection is that you referred to something as being put all in one place and I gather that what you were talking about is the systems interaction -- systems interaction function. And the question that I was trying to ask was intended to be a very straightforward one.

How would that function be accomplished vis-a-vis say the 21 technical review branches that we now have? Would they continue to do their thing to perform their function and there would be some other organizational unit created to deal with systems interaction, as I commented on, in trying to restate the question?

Perhaps it's premature to ask the question, because I gather that what the final implementation of the systems interaction study will be is maybe too far off at the present time, or maybe there are too many unknowns involved.

Do you understand what I am trying to ask?

A Yes, I understand. When we first defined our problem two years ago, we did speculate that there were two ways to resolve this problem:

One was to better define the areas of responsibility of the 21 or so technical organizations and the review plan, so that their overlapping and interrelationships with other technical areas was better defined.

The other way to solve it may be to have a separate group who would take an overview of the power plant.

Now two years later, when we're beginning to get some of the results out of our work at Sandia, I would speculate that we are going to resolve that by better definition of the review plan, where it's needed.

It now beings to appear that whoever the major parties were in writing up the review plan, they had a very astute -- I don't know what word to use -- perception of how to break up the review.

It appears that if there are lapses in the review plan, they are not that significant, and they can be easily modified.

That's my present opinion about where we're going. And like I say, our preliminary results appear to indicate that system interactions can be handled by modifications to the review plan, and not very significant modifications, either.

Q You mentioned earlier when you were -- the date on which you were appointed the task manager for this project, that was some time in 1977?

A Yes.

Q And the effort was initiated when, in '77 or '78?

A Well, as soon as I found out about this, by the letters of June 1977 that we talked about a few minutes ago, and the first inkling I had that the NRC was going to do something active was when I was called in to Dominic Vasallo's office and informed that I was under consideration to be the task manager, and wanted to know whether I would agree to be the task manager.

Q To the best of your recollection, and also to the extent that you may have been involved and aware, what accounted for the -- what, several years, perhaps four years delay from the time that the Advisory Committee on Reactor Safeguards raised the issue until an initiative was initiated by the Nuclear Regulatory Commission? Do you have any information or understanding?

Perhaps the question should not be addressed to you.

A Well, yes, it was a concern to me when I was first assigned as task manager, was why did it take three years to get something going on this.

Q Right.

A My answers to that kind of a question were discussions with Dick DeYoung. At that time in 1977, Dick DeYoung was appointed to be the lead supervisor for systems interaction.

Apparently in my discussions with Dick DeYoung, we had attempted several times to discuss this with the ACRS to convince them that it really wasn't a problem that required any more action than we were already taking.

We felt that our normal processes of review and inspections of plants were sufficient to flush up system interactions, or those areas of concern.

How many times we went down to the ACRS and what the nature of all that discussion was, I'm not aware of, except that we apparently had gone to them a few times to try to convince them that it really wasn't a matter of the utmost concern here.

Q As far as you are aware, during this period of several years, that is between 1974 and '77, were the Commissioners involved in the issuing -- in other words, was there any briefing of the Commissioners, to your knowledge, about the issue of systems interaction?

A No, I don't believe so.

Q To your knowledge, were there any directions from the Commissioners?

A No.

Q To your knowledge, were the Commissioners even informed that the ACRS had raised the issue of systems interaction?

A No, I can't say that-- I don't have any evidence that the Commissioners were ever involved in this. In a lot of comments the ACRS, in that collection of letters, referred to interactions with regard to standard plant, and the Staff and Applicants for standard plants had already taken a considerable number of actions that could be classified as response to concerns about system interactions.

These came under different names called interfaces and in particular were concerned with coordinating the technical work and design between the two parties generally to a standard plant, that is the nuclear steam system supplier and the balance-of-plant designer.

So I could say that there was some activity over that period of years from 1974 till 1977 that in a loose way could be defined as system interactions.

Q What, the concern about interfaces and standardization area?

A Yes, that was a very broad look at the coordinating the efforts of two very large groups. That is the designer of the nuclear system and the designer of the balance of plant. But in effect it probes,that kind of interface study probes at a lot of potential interactions.

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Q Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

We are on the record now.

THE WITNESS: Go off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

BY MR. PARLER:

Q What is your understanding of some of the initiating reasons for the Advisory Committee on Reactor Safeguards' concerns in the systems interaction area? You have mentioned earlier some of the broad concerns that were raised in regard to interfaces between the nuclear steam supply system and the balance of plant, and the review of standardized designs. Do you have any comment on that?

A Yes, that's because that collection of letters that we had referred to earlier, a significant number of those do mention the interface problem between standard plant designers, significant number of them are related there.

Let me just extract one as an example. And these were marked, these particular passages are marked, for example, RESAR 3-S. I'm reading now from one of the letters from the ACRS.

Q ACRS letter?

A Yes.

Q Why don't you give the date of it?

A That's the July 1., 1976 letter.

Q That's one of the 15, right?

A Yes, it's one of the 15. July 14, 1976.

Q That's all right, go ahead.

A That's a letter to Mr. Rowden.

Q Right.

A From Mr. Moeller, Dr. Moeller of the ACRS.

Q All right, that's good enough. Go ahead.

A In which he states -- the letter states that RESAR 3-S provides for those safety-related interface requirements that are essential to designing the balance of plant to be consistent with the assumptions used in the accident analyses.

He says since the utility applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related design requirements have been met, these matters will be reviewed in more detail with the utility applicants on a case-by-case basis.

The committee recommends that during design, procurement, construction and start-up, timely and appropriate interdisciplinary systems analysis be carried out to assure complete functional capability -- I'm sorry, functional compatibility across each interface for the entire spectrum of anticipated operations, and postulated design basis accident conditions.

Q You said that this language you just read from was marked. What does that mean, incidentally? I realize you are talking about a bracketed mark in the margins. Was that something that the ACRS uses to highlight a particular point, or what is it?

A No, I believe that's where Mr. Fraley marked each of the letters to indicate what portion of the ACRS report he thought referred to system interaction, and the words "interface" and "interactions" got intermixed.

Q All right.

What is the present status of the systems interaction issue as far as the Staff is concerned?

First of all, I gather that it is a generic item; is that rignt?

A Yes, it is.

Q Is it considered to be a generic unresolved safety item?

A That is its category now, yes. The Commissioners placed it in that category.

Q I gather from what you have said that during the period between 1974 and 1977 when the dialogue between certain Staff members and the ACRS was taking place about the need to conduct a study in the systems interaction area, that at least during those years, this issue or the issue of systems interaction was not deemed to be an unresolves safety item.

Is that correct?

A   I'd have to interpret that. I have no personal knowledge of how the Staff viewed that problem until I became involved in it in 1977.

Q   Do you say that the Commission -- I assume you mean the Commissioners -- placed this issue or the issue of systems interaction in the category of an unresolved safety item? Is that right? Or do you know?

A   Well, let me try to go back here. Maybe I have my words mixed up a little bit.

We, the Staff, included system interaction as one of approximately 40 Category A generic tasks, and then when we made our report to the Commissioners, our recommendation of whether these should be considered resolved -- unresolved issues as compared to generic matters that we would pursue, but not in the category of an unresolved safety issue. The Commissioners took a different viewpoint and deemed that system interaction, because of its broad implications, should be considered unresolved safety issue, at least until we had completed the first phase of our work, and then there would be another judgment made as to whether it would be continued as an unresolved issue or dropped from that category.

Q   Do you have that report that you referred to, the Commission, with you?

A No, I don't have it with me.

Q I assume that you are talking about a Staff paper to the Commission, at least it's one that I have had referred to me. It's SECY 78-616 of November the 27th, 1978, and on page 10 of an attachment to that paper, there is a reference to the A-17 issue, that is systems interactions and nuclear power plants, and in the paragraph describing that issue, there are the words, "This issue has been determined not to qualify as an unresolved safety issue because it does not represent a possible major reduction in the degree of protection to the public health and safety."

And the words on that Staff paper continue to say, that, "We," that is the Staff, "believe the likely interactions that have significant consequences are being addressed by both the designers and the Staff in its review, and that Task A-17 will confirm this judgment. Accordingly, Task A-17, systems interactions, does not qualify as an unresolved safety issue."

Again those words are from the Staff paper, SECY 78-616, dated November 27th, 1978 from Harold R. Denton, Director of the Office of Nuclear Reactor Regulation, to the Commissioners.

Now that would appear to you to be report from the Staff to the Commissioners that you referred to earlier?

A Yes, that's the report.

Q I gather from what you have testified to that when the Commission reviewed the paper that I referred to, presumably they did not agree and they decided that this particular issue should be considered an unresolved safety issue; is that correct?

A That's correct. The Commission did not agree with that.

Q I also understand that in connection with the Staff's categorization and descriptions of these generic items, including systems interactions, which is the only one that we are concerned with this morning, that the probabilistic analysis staff was also asked to review the issue and comment on it. Is that right?

A Yes, they were.

Q Do you happen to recall, after you take the time to refresh your recollection, what the probabilistic staff's analysis of the issue was from the standpoint of its safety significance?

A I believe that the research staff categorized systems interaction as having a -- being a substantial contributor a potential substantial contributor to -- I don't know how I'd characterize it -- core damage or safety, let's say substantial contributor to nuclear safety problems.

But in discussions with them, it at least was my opinion that their definition of system interactions was somewhat different than my definition of system interactions. But accepting the researcher's definition of system interaction, I could agree that of the residue or balance, you might say, of risk, system interactions probably did predominate that balance.

That is to say that whatever small amount of risk is still left, system interaction, by Research's definition, predominated the risk.

Q Is that -- I see.

In an attachment, I believe it's page I-11 of the Staff paper that I referred to earlier, in the Research's write-up of this issue, the A-17 systems interaction issue, they say that, among other things, this:

"If the Task Action Plan proposed for this program is conducted properly, it is expected that the results will show that systems interaction dominates accident risks as they did in the reactor safety study."

Now are those the words that you were just explaining? To a layman, it isn't entirely clear what these words mean when they refer to systems interactions dominating accident risks. Could you comment on that, please?

A I'm not too sure what they mean, either. If you take -- all I can speculate is,if that is permitted in here, to speculate --

Q Of course. Those are not your words. Those are the words of the probabilistic staff in the Division of Research in reviewing this background material for this occasion. Those are words that were not entirely clear to me, so I'm asking you only in your capacity as the task manager. Perhaps you haven't had the occasion in the past even to reflect on those words.

So it's just your best judgment on what you think they meant.

A In the sense that we are looking and define system interaction, I couldn't agree with the statement that those kind of interactions dominate the risk. If I look at what I believe to be Research's definition, Research staff's definition of system interaction, then I'd have to say, well, whatever residual or whatever small amount of risk there is in nuclear power plant is probably dominated by system interaction, and that is that system interactions are going to be the contributors to whatever small amount of risk there is.

But when I made a response to that concern, which led to the Staff's position that systems interaction was not an unresolved safety issue, the position that I took and the position that I wrote up was that if you were to consider all system interactions that are possible, yes, I agree with Research that they dominate the risk.

But if you consider the -- if you consider the system interactions that are left after the Staff has conducted its review, then I would have to say that we fully expect that those kind of interactions probably do not dominate the risk at all.

Let me try to explain that. For example, Research might put at the top of the list of systems interactions turbine missiles. And I would agree that if a turbine were to fly apart and spew its missiles all over in almost all directions, it has a potential of interacting with many other systems.

It might destroy or damage a significant number of other systems in the strike zone. That is if the barriers weren't sufficient and the missile had enough velocity and energy and all sorts of things like that.

But if we look at -- if we look at the review of power plants, both the design of the power plant and our review of those power plants, I would have to conclude examination of the review plan indicates that that interaction is not left undetected.

In other words, we make a specific detailed review of turbine missiles and plant alignment. So I would have to say then that after the plant design and our review have been conducted, the risk now from interactions due to turbine missiles has literally vanished.

You might take another example. For example, floods are certainly a potential for causing a lot of undesirable system interactions, but the Staff and Applicant specifically conduct a considerable amount of analysis with regard to floods. So that interaction disappears.

I guess what I'm trying to say, that if you take all of the possible system interactions and compare all of these against our review and our criteria, we would find literally all of these are accounted for in our review.

So what we are looking for in this generic task that we are talking about now, what we are looking for is stuff that has escaped our attention.

If you look at those system interactions, then I'd have to go back to our original statement. We don't believe that those are significant and that they pose an unresolved safety issue. That's -- however, I cannot -- I don't mean to imply here that the Staff disagrees with the Commission action in placing system interaction as an unresolved safety issue, because I think that we would agree then with the Commission viewpoint that since the -- since the problem has such broad implications or such concern, then I would have to agree to put it as an unresolved safety issue, at least until we make our first confirmation.

Q All right. You mentioned --

A Off the record.

[Discussion off the record.]

MR. PARLER: On the record.

BY MR. PARLER:

Q You mentioned earlier that the Commission decided to include the task A-17 to the list of unresolved safety items.

For the record at this point, I gather that that specific action by the Commission and the earlier Staff paper that I mentioned, SECY 78-616, is reflected in Mr. Samuel Chilk's, the Secretary to the Commission, memorandum of December the 13th, 1978. And the recommendations of that memorandum are in another Staff paper from Mr. Denton to the Commissioners that is identified as SECY 78-616A, dated December 28th, 1978.

That paper that I just mentioned has the following -- or a write-up on the issue A-17 which I just handed to you, Mr. Angelo.

As far as you are aware, these references that I have given and the dates are consistent with your understanding of the directions that the Staff received from the Commissioners on this issue; is that correct?

A Yes, that is correct.

Q And these papers were concerned with the preparation of an annual report to the Congress on unresolved safety issues; is that right?

A Yes.

Q Now this excerpt that I handed you, Mr. Angelo, from the SECY 78-616A, it does describe the background of the task A-17 which you have already covered in your testimony, and this excerpt at pages 25 and 26 to the Staff paper 78-616A emphasizes, as you have already in your testimony, that adverse effects might occur because designers might not, for example, assure that redundancy and independence of safety systems are provided under all conditions of operations where redundancy and independence is required, because the functionalities might not be adequately coordinated.

Simply stated, the left hand may not know or understand what the right hand is doing in all cases where it is necessary for the hands to be coordinated. But nevertheless I understand that it is your understanding or your view that even though this issue on systems interactions is deemed to be an unresolved safety issue, that pending the completion of work on the task, that what the Staff is doing in its review of individual applications is adequate. Is that right, sir?

A Yes.

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Q I gather that the reason for that essentially is stated on page 25 that was appended to the SECY 78-616A; is that right?

A Yes, that is right. That statement -- the statement is that the NRC Staff believes that its current review procedures and safety criteria provide reasonable assurance that an acceptable level of redundancy and independence is provided for systems that are required for safety.

I think our results so far seem to still affirm that belief.

Q You mean the results of the Sandia study and the other ongoing studies?

A Yes. I don't want to prejudge what will finally come out of this, but that is my indication, we have made every effort.

I might add in conducting this task to keep Sandia as independent as possible from this, you might say, judgment that present procedures and criteria provide reasonable assurance, we have left them to conduct this task in a way so as not to be prejudiced by what the Staff may conclude or what the Staff may feel about it.

And I think I feel quite happy and satisfied that Sandia does conduct themselves that way. They have maintained an independence of spirit in doing this job.

Q When do you expect to have the results of the ongoing task completed? If the answer involves certain phases, why don't you so indicate?

What I'm getting at is, I would assume that the study would involve an analysis or report and the report is one phase and then there would be another phase having to do with the implementation of the results of the study.

Would you comment on that, please?

A Well, that is the way we perceived the task to be two years ago when we started out. We said that the first phase would be an investigation of system interactions currently, and the second phase would be implement whatever it was that we discovered in phase one.

Currently we expect to finish phase one, the actual work will be finished by the end of 1979, and that would leave us a couple of months to put it together in a form that we can communicate our findings to all the parties, including the general public.

It would allow at least until the end of March 1980 to do that.

In the meantime, we are going to be thinking about what it is we are going to do for implementation in what we call phase two, but I have a suspicion that phase two implementation is not going to be so great. I would imagine that what's going to come out of this is follow-on work to phase one, in which we would investigate some areas that we are not now covering in phase one. But that's pure speculation on my part at this time.

We really are reserving our judgment on what to do in phase two, whether we do any follow-on studies to phase one or implement phase one, until we have reached some review and consensus by the technical people involved in the NRC.

I really believe that the limited scope we are looking at here is really not going to pose a large problem of implementation. I think the larger problem is whether based on results of what we get out of phase one, should we do additional studies like operator errors, design errors, and installation errors. I think that's the --

Q How about the classification of equipment and components and perhaps even systems as safety or nonsafety grade? How does that bear on the study?

A In the study we have made no distinction between grades of equipment, safety grade or nonsafety grade. Instead we have defined safety functions that have to be performed and then we went out --

Q For the entire plant?

A Yes, for the entire plant. And then we go out and look at all the equipment that can perform that function, even if it is nonsafety grade equipment.

For example, core decay heat can be removed by systems in the power plant that are nonsafety grade, as well as by systems that are safety grade. So we are essentially interested in whether there is any interaction possible among all these systems.

Q And your interest in that regard, I gather, is not limited by the application of the single failure criteria that we -- that the NRC follows as a part of its regulatory philosophy?

A No -- well, I'm trying to get the sense of your question. We are all aware of the single failure criteria as we do our study, but that doesn't influence the way we do our study. We're looking at all the ways in which equipment can be faulted in various combinations.

What we are finding generally is that it takes more than a single failure to cause a safety problem. I guess we knew that before we started the study.

Q What I'm trying to ask is whether your study is being bounded by principles that are already a part of the regulatory practice.

One of them would be the single failure criterion. Another is the one that I have mentioned, the classification of equipment as safety grade or nonsafety grade.

Another would be that we don't look at accidents beyond the design basis accident. That's the thrust of my question.

A Well, I think I have already answered that one about the classification of equipment.

With regard to single failure criterion, we are -- let me try to explain something more that I probably should have explained earlier, and that is that once we've identified the combination of equipment that could negate or degrade the performance of the plant in performing a safety function, we then go back to the review plan to find out whether that particular kind of interaction is discussed in the review plan.

If it is, our task ends there for that particular interaction, to say that it hasn't been overlooked, that particular interaction has not been overlooked in the Staff's review or in their design criteria.

If we find an interaction that is not addressed in the standard review plan or perhaps not addressed satisfactorily, then we would have to decide whether to include it in the review plan.

In that regard, we are not -- we disregard -- or not disregard, we are not bounded by the single failure criteria. For example, we have made a decision to carry more than the single event in the system interaction.

In fact, we are carrying as many as three events, independent events. Even though we decide that these independent events are not caused by system interaction, we still retain them in the study and most of the three-event things we won't go back and look at the review plan with regard to those three events, because we know ahead of time that the review plan doesn't prohibit three independent events.

In fact, most of the review plan talks about single failure as the criteria. No single failure shall prevent safety function.

So, in a sense you might say that we probably don't have to go back and check our review plan against criteria in any more than a single event, because -- but there are a few cases where the review plan does talk about more than one event, and that is in particular in the auxiliary feedwater system.

Our present review plan requires that the system be designed for -- for example, no dependence on AC power. This presumes that we have suffered more than one dependent failure and have lost both offsite and onsite power, for example.

So we will go back and check the review plan for two events; probably will not check it for three events. I'm not sure whether I'm answering your question or not, but in that sense we are not bounded, we are not bounding the problem by the single failure criteria in the sense that we are going to look at and evaluate more than one event.

We will look at two or three events.

Q How do you deal on the study with the role of the architect/engineers? Are you assuming there is some sort of hypothetical plan or what?

A Yes, we have tried to -- not hypothetical plan. We have taken the study in two categories, really. We do an analysis on a generic basis, recognizing that somewhere along the line the interactions can be dictated by a specific plan arrangement --

Q What do you mean by a generic basis? Do you have some sort of a configuration that you assume, or what?

A No, it's really safety functions that are generic in a plant. For example, removal of core decay heat is a generic safety function to any nuclear plant, and the systems that do that job are pretty much generic in their general configuration.

But then you become plant-specific in the physical arrangement of that equipment and the way it might be controlled. So to take care of that, we have carried the study to an exemplary plant, in this case we've used Watts Bar, but the vehicle is mostly -- the vehicle of an exemplary plant is mostly to demonstrate that our technique is a workable technique and can be applied to a specific plant.

Q You say Watts Bar is the exemplary plant?

A Exemplary facility, yes.

Q Is that why you and others went down there last Friday to collect detailed information on location and operating characteristics of plant equipment needed for the evaluation of fault trees for task No. A-17?

A Yes, that's the purpose. It's a demonstration bed, if you want to say it, that we can actually take a generic problem like this and apply it to a specific plant and come up with a workable result. It isn't meant to imply that you can't -- that you have to go to a specific facility to solve a generic problem.

Q The work that you are doing, I would assume that it might make a difference whether one is talking about some future plant that is being built for a utility, either public or private, that would have a considerable amount of in-house engineering capability, and one that doesn't.

In other words, if a company contracts for architect/engineering services, as contrasted to having those services performed in-house, that would have a bearing on the issue of systems interactions, would it not?

A Well, yes. Where it is, the bearing, though, that we are attempting to find out where in the review plan we may have overlooked or not recognized those kinds of things that you infer in your question. It isn't that; for example, we are trying to point where the reviewer or the review plan should look for an interaction.

Q The obvious points I would understand, but I gather from some things that I have read that one of the concerns in the area of systems interaction are the hidden things, the things that people have not thought of, and I don't know -- is that your understanding also?

A Essentially, yes. We are attempting to find out where we have lapses or where we have overlooked items in our reviews that could contribute to system interaction.

Q What I would like to ask you is, as a part of this study, are you collecting data to show what the experience has been in that regard? That is, hidden things, hidden issues, because of systems interactions, and look to see how the data is, if you have a turnkey job, how the data is; if you have a utility that's involved that has a considerable amount of in-house engineering and architect/engineering capability; and how the situation is if you have the opposite of a turnkey job? That is a small utility which depends very heavily on outside engineering and architect/engineering support? Or is all of this sort of stuff irrelevant as far as you are concerned?

A Well, I think as far as safety is concerned, the way we review safety and the way we apply criteria, that is irrelevant. We make no distinctions among who the parties are.

Q But we may not find everything. There may be the potential for hidden issues -- I guess commonly in the vernacular referred to as boo-boos by architect/engineers, et cetera.

A That may be true. There may be more or less potential, depending on the parties involved.

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But it wouldn't affect the way we are doing system interaction or the way we would identify things that slip through the cracks. We are still trying to confirm the roadmap that the NRC uses in its review of nuclear plants, regardless of who the parties are involved in the design or roadmap. It's still the same roadmap, and what we are attempting to do here is make sure the roadmap has not left out some of the pitfalls.

Q That roadmap is the standard review plan?

A That roadmap is the standard review plan.

Q The work that Sandia Laboratories is doing in the area that you are the task manager for, is that the only work that Sandia Laboratories is doing with regard to the effectiveness of the standard review plan that you are aware of, or is there some other work that they are doing?

A No, they are doing other work for Research related to the standard review plan.

Q Do you know how that work relates to the task A-17?

A Well, somewhat. I'm not sure if all of it's detailed, but we do know that we are getting a considerable amount of benefits from that work that they are doing in the standard review plan, in the sense that some of the same people are involved in both tasks. So we are able to gain the benefits of the very detailed work that Sandia is doing on the standard review plan.

What it does is, it means that the people engaged at Sandia in doing system interaction studies can much more quickly now take the results of system interaction and compare them to the standard review plan, simply because they have a very detailed knowledge of the standard review plan, probably more detailed than any single group of people I could name. So we gain a lot of benefit from them, although the thrust of the work is slightly different. If you -- the thrust of what Sandia is doing on the standard review plan for Research is slightly different than what we are doing in system interaction.

Q Maybe I've missed your point, but what is that slightly different thrust, without any great detail? Could you state that for the record? I gather that they are looking at the effectiveness of the standard review plan for Research, whatever that means, but maybe you don't know.

In any event --

MR. PARLER: Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

BY MR. PARLER:

Q In any event, whatever the probabilistic analysis section in Research has Sandia doing with regard to their Sandia review of the standard review plan, and although that company's familiarity with the standard review plan may be helpful, their research work with Sandia, as far as you are aware, is not directly related to the systems interaction effort. Is that right? That's a separate one?

A No, it is not directly related.

Q How about with regard to the use of operating experience in Licensee Event Reports in connection with this study? What's being done there? What use is being given to that information, operating experience?

A The use we have made of that experience is to make sure that we understand the kind of interactions that are possible in the plant and the time of characteristics that equipment has that would lead to interactions.

For example, we can discern from Licensee Event Reports which kind of equipment has the characteristics of failures due to lubrication or failures due to cooling, failures due to power actuation, and that sort of thing.

Q Who is doing that work?

A Sandia. Well, we really relied on the Zion interaction study, and made an analysis. Sandia has made an analysis of the results of the Zion study.

Q Does the Zion study --

A That took 9000 Licensee Event Reports and in a sense it boiled these 9000 events down to about 260 that were potential interactions, and then made another big step down to about 70.

Now to go from 9000 to about 270 was quite a major step. The next bunch down to about 70 was to eliminate things like release from radioactive waste systems and operator errors which we eliminated from our task.

Q Have the results of the Zion study been published any place?

A Yes, they are published in a report by Commonwealth Edison called "The Zion Station Study." I'm not sure of the exact title of it.

Q That was the report that Commonwealth Edison provided to the Advisory Committee on Reactor Safeguards?

A Provided it to the NRC Staff, to the Division of Operating Reactors, and that report was made available to the ACRS.

Q About how long ago was that study put out? Do you have any idea? Recently?

A I believe it was published in June of 1978. June of '78.

Q And those were the Licensee Event Reports on the operating experience that Sandia is basically relying on for the systems interaction study, or are they going beyond that, or do you know?

A Well, I don't know if you'd say relying on it, no. We have made very limited use of it. The only use we have made of the Zion study is to make sure we haven't missed some sort of interactive characteristic among plants.

Q What I'm getting at is this: I gather that, as you have said, there is a Zion study which took a look at Licensee Event Reports. It is also my understanding that a subcommittee of the Advisory Committee on Reactor Safeguards is working on a report on Licensee Event Reports. I'm trying to find out whether in connection with the A-17 study, there is yet another study looking at Licensee Event Reports.

I'm interested in such studies because certainly after March 28th, 1979, questions have been asked about the adequacy of the evaluation of certain Licensee Event Reports for other purposes that have nothing necessarily to do with the topic that we are discussing this morning.

So are there any other studies of Licensee Event Reports that you are aware of, that are being undertaken in connection with the project A-17?

A No.

Could I add something?

Q Oh, yes.

A I'd have to say that task A-17 really does not purport or try to convey that the task is going to do any discipline study of Licensee Event Reports. We really aren't doing that at all.

Q In connection with this study, do you know whether Sandia or anyone else is looking at how other countries approach the issue of systems interactions?

A No, I'm not aware of it. I've only had very brief discussions with the Swiss delegation, and they wanted to know what we were doing in system interactions.

I don't believe I can say I know of anything that other countries are doing specifically.

Q I suppose what I was trying to ask you is in connection with our study, the systems interaction study, was such an attempt made or is it being made -- I gather as far as you are aware, no?

A No.

Q I will hand you a document -- off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

BY MR. PARLER:

Q I'm going to hand you a document, or I have handed you a document from C. Michaelson. "C" is the first initial, Michaelson, M-i-c-h-a-e-l-s-o-n, to M. Bender, who is the Chairman of the Plant Arrangement Subcommittee of the Advisory Committee on Reactor Safeguards.

This document is dated January the 17th, 1979, subject, remarks concerning subcommittee meeting on plant arrangements, October the 25th, 1978.

I'll mark this document for identification as Exhibit 1068.

[The document referred to was marked Exhibit 1068 for identification.]

BY MR. PARLER:

Q I gather, Mr. Angelo, that you received a copy of this document earlier?

A Yes, I have received a copy of this from ACRS representative and from one other source. I'm well aware of the document.

Q Are there -- I gather that this document, the Exhibit 1068, is a report by Mr. Michaelson, a consultant to the ACRS, to Mr. Bender, a member of the ACRS, who presumably is head of the Subcommittee that, among other things, is concerned with systems interaction? Is that right, sir?

A Yes.

Q Is there anything about Mr. Michaelson's comments about the systems interaction study that you would like to address yourself to?

If not, I will just include this document as an exhibit and Mr. Michaelson's comments are there for anybody that wants to read his memorandum to Mr. Bender.

A Well, we could, of course, take many hours to go through a point-by-point discussion of all of these, but I don't think that would serve much purpose now. I think that I'd make a general statement that we are aware of all of the concerns expressed by Mr. Michaelson and we make the statement that we were probably not accounting for most of his concerns. We are probably accounting for only a small percent of his concerns.

Q I suppose a question that I would like to ask is whether the uncertainties as to what the ACRS wanted done in this area, that as I understand it existed during the period 1974 to 1977, when the task A-17 was initiated, whether those uncertainties between the Staff's understanding of what the ACRS wants and the ACRS' understanding of what it wants, whether those things have been resolved.

Off the record.

[Discussion off the record.]

MR. PARLER: Back on the record.

THE WITNESS: I think I know what ACRS wants. I believe that I have a pretty good understanding of what the ACRS wants, and I also understand that we are not going to give them more than 1 percent of what they're asking.

BY MR. PARLER:

Q You said 1 percent?

A I would estimate that we are going to answer 1 percent of their concerns with this study. But we hope in the demonstration of this 1 percent that the other concerns of the 99 percent will have been, I suppose, adequately resolved. I'm not sure. I inferred earlier that the most likely outcome of this task would not be implementation in phase two, but it would be follow-on studies to include some of these elements that we know we don't have in our present study.

Q I realize that we are talking very generally with these percentages here; in any event, I gather from what you just said, that a large percentage of your understanding of what the ACRS wanted accomplished in this study probably is not going to be accomplished?

A That's right.

Q Now what is --

A At least not in this first phase that we are going through now.

Q Why is that? Is that because of the things that you mentioned at the outset of your testimony, a couple of hours ago that were excluded from the study, or is there something else involved? I'm not clear on that.

Generally speaking, I realize that it would take a lot of time to go through all the details.

A I think that we limited this study for several reasons. The principal reason was that we thought we had a feeling in 1977 that the principal concern of the ACRS was the physical configuration of plants in their -- as to whether they met the presumed redundancy and independence of plant systems, safety systems.

In that regard, I think our study takes care of virtually all of that concern. In other words, our study is really -- concentrates on the physical arrangement of the plant from the point of view of the independence and redundancy of safety systems.

The other reason we limited it was because we were faced with the task of trying to demonstrate whether we even had a viable or believable approach to how to resolve the question of system interaction.

There was a concern as to whether we could even develop a method of doing it, as the subject -- the question was the subject is so vast and included so many things, that maybe there wasn't any practical way of resolving that, and we had to demonstrate and I think we are going to succeed in demonstrating that there is a practical, viable, feasible method of solving the problem of system interactions on a global scale.

Q Well, is what you are telling me and stating for the record that if there is the kind of demonstration from this study that you've just covered, that in your judgment that should take care of a large part of what would now appear to be the difference between what this study is doing and what presumably the ACRS wanted done? Is it just a question of placing realistic bounds on the study, or is it something else?

A That was the real question, placing a realistic bound on it, so that you could -- so that you could demonstrate that at least within this area, we can address the question of system interaction.

I don't know maybe I'm naive. I had the feeling that we might be able to resolve all of the ACRS' concerns with what we will accomplish by the end of this year. I don't really believe that we are going to do that. However, I am still confident that you can take the question of system interaction and place it in a different category. In other words, not the unresolved safety issue that it is, but some lesser category of concern.

I think we -- I'm hopeful that we can at least demonstrate -- if we demonstrated anything, that we can demonstrate that.

We have taken a very disciplined and coordinated look at systems in their performance of safety, and by what we find out, we would be able to categorize those as still an resolved safety issue or as an issue that still bears some further look at, but isn't in the category of unresolved.

I guess that would be the most I would hope to get out of this task by the end of this year. I really can't speculate whether we are going -- what the ACRS will think it is.

Q I wasn't asking you to speculate in that regard, and quite properly you shouldn't. What I was trying to find out from you is the reasons for the apparent different views and the approach that should be taken.

I gather that you have commented on that. Maybe the ACRS anticipates a much broader study than the Staff here believes is feasible or is realistic. Isn't that what you have been saying?

A Yes. I would have to characterize it that way.

Q And how the matter will eventually come out, no one can say at this point; right?

A That's right. It can't be said at this time how it will come out. I -- the question really is whether the physical configuration of a plant contributes to system interactions as much as the human element, you might say, the operator contributes to it.

You might say even though the designer has designed a plant in such a way that the systems are independent and redundant, and the things that occur in one system don't affect the other system, there is still a potential that the operator can couple the systems.

You can look at the operator as a free, roving system all by himself. That is, he can discern things in the plant and he can manipulate systems in the way that the designer never put into that system.

The designer, for example, may never have designed a plant and there isn't a single thing in that power plant that would shut off two safety systems at the same time, but the operator can do it.

The question really came on system interaction, where should you put the biggest emphasis? Should you put it on the physical configuration of the plant, or should you put it on the human element, the operator or the man who installs the equipment wrong? He took the drawing by the designer, that was an absolutely perfect drawing, but he put the thing in upside down and nobody discerned it.

Q And the A-17 study is putting the emphasis where, on the physical arrangement?

A On the physical arrangement. However, we are trying to emphasize to people that by looking at the physical configuration, we can find the places where the design may be sensitive to the operato s errors.

In other words, we will be able, hopefully, to discern where the operator is more likely to make errors.

Q Do you have anything else to add about the systems interactions task? Anything that you would like to add yourself, either because a question has not been asked, or because of the way a question has been asked?

A Well, I would like to add one thing that I think is important for everybody to understand and realize, that regardless of any beliefs or notions that the Staff may have about the adequacy of its reviews or the adequacy of designs, or whether we have overlooked important areas or not, that what Sandia is doing has been not prejudiced by this at all.

I have made every effort, and I think the Staff has, too, to make sure that Sandia is not prejudiced in the performance of this task; that whatever they come out with isn't prejudged or influenced at all by our conclusions or statements that we came to before we started the task, that said that, you know, we didn't believe system interaction was an unresolved safety issue.

Q All right. Anything else?

A That's all.

Q Now I'd like to show you a document which, as I understand it, really was the initiator at the Staff level of this particular task action A-17. It's a memorandum from Roger S. Boyd who at the time was a Director of the Division of Project Management, memorandum from Mr. Boyd to Edson G. Case, who then was the Acting Director of the Office of Nuclear Reactor Regulations.

This document is dated May 1978. The precise date is not legible. The subject, submittal of revised task action plan.

I have given Mr. Angelo a copy of this document.

Is my understanding of this document essentially correct, that this is a document that established or proposed to establish a task action plan for task action A-17, for which you were the task manager?

A Well, yes, there were some plan -- there was a basic plan that was generated before this date. This particular revision reflects Sandia's involvement in the task. Up till then we hadn't decided who was going to do the task.

Q I see.

There was an earlier -- an earlier version; right?

A Yes. Oh, yeah. And this has been revised since then, even. There's a later one, but it hasn't been approved.

Q Is the substance of this memorandum correct, as far as its description of the project and the leading actors in the project, et cetera?

A Yes, the substance is correct.

Q All right. Now you say there is still another revision which has not been approved? Is that what you just said?

A Yes.

Q Would that make substantive changes to the task?

A Not substantive, no. It changes only slightly. The participants, for example, in the NRC. And it does recognize tha the work at Oak Ridge has not -- is not going forward.

Q But as far as the objectives of the task are concerned and the basic approach, that is still the same as described in this document which I just handed to you? Is that right?

Take your time and look at it.

A That is true. That is right. Later versions of this, which are still going through to be approved, detail a little bit more the approach that Sandia is using to resolve the task, but the substance is still the same, except that later revisions describe a little bit more the work that Sandia will do in a little more detail.

I might add it's a little difficult to keep the task action plan up to date. We've had additional revisions because since Three Mile Island, all the unresolved safety issues are now in a different organization, and different personnel are involved in some of these tasks.

I would have to state, though, the sum and substance of what we are doing has not changed.

Q I want to read for the record an excerpt from page 2 of Exhibit 1069. This is in the introduction to the task action plan in task No. A-17, and is under a section entitled "Problem Description."

The language that I want to quote for the record from the exhibit, 1069, is as follows:

"Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the inter-disciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of our regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

"Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems."

That's the end of the quote.

At the time this document was written, May of '78, I gather from what has been said, that as far as the Staff is concerned, or was concerned at the time, the issue of systems interaction was not an unresolved safety issue. Is that right?

A Yes.

[The document referred to was marked Exhibit 1069 for identification.]

BY MR. PARLER:

Q That decision was made by the Commission in December of '78, as we earlier discussed; right?

A Yes.

Q All right. Now, do you have anything else about systems interaction or task A-17 before we take a small break and move on to something else?

A No.

Q How about Mr. Cox? Do you have any questions?

MR. COX: No.

MR. PARLER: Do youall mind if we take about a five or so minute break?

[Recess.]

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BY MR. PARLER:

Q Now it's my understanding that you don't have any other comments or information to provide on systems interaction; is that right?

A That's right.

Q Now we'll move ahead briefly on a completely separate topic. It is my understanding, Mr. Angelo, that some time ago, as a matter of fact, almost three years ago, you were appointed as a member of a group to review and compile information on what principal architectural and engineering criteria mean, and to develop decisional criteria to judge postconstruction permit design changes; is that correct, sir?

A Yes, that is correct.

Q The document that is involved, I would like to mark for identification as Exhibit 1070. This exhibit is a memorandum from R. S. Boyd, who was then the Director of Division of Project Management, to Ben C. Rusche, who was then the Director, Office of Nuclear Reactor Regulation and the date of the memorandum is January the 25th, 1977.

[The document referred to was marked Exhibit 1070 for identification.]

MR. PARLER: Exhibit 1070 refers to another memorandum, which I will mark for identification as Exhibit 1071. Exhibit 1071 is a memorandum from Mr. Rusche to Mr. Boyd, dated January the 24th, 1977, subject, post-CP application amendments.

[The document referred to was marked Exhibit 1071 for identification.]

MR. PARLER: Now as a result of your assignment to the task force on post-CP applications amendments, I gather, Mr. Angelo, that you worked with Richard DeYoung and Bill Kane on the assignment, at least to some extent; is that correct, sir?

THE WITNESS: Yes, that is correct.

BY MR. PARLER:

Q Were you involved in the assignment throughout the w. .k of the task force during the early months of 1977 to the best of your recollection?

A Yes, I was involved in the early work. I was reassigned before it was completed.

Q So the recommended report of that task group on post-CP applications amendments, you were not on the task force at the time of that report; right?

A I don't believe so. However, it doesn't look changed much from what it was while we were working.

Q So you are generally familiar with the report of the task force?

A Yes.

Q Although you may not have been around at the final date of its issuance; right?

A That's right.

Q The document that I have given you, to the best of your recollection, does that appear to be, at least in substance, the report of that task group?

A Yes.

Q All right.

I'd like to mark for identification as Exhibit 1072 a document from Roger S. Boyd, then the Director of Division of Project Management, to E. G. Case, subject, post-CP application amendments. March 7th, '77 is the date. And this exhibit, 1072, does forward to Mr. Case the report of the task group which studied the matter of identifying the principal architectural and engineering criteria.

[The document referred to was marked Exhibit 1072 for identification.]

MR. PARLER: In addition to this report, Exhibit 1072 also indicates that the report has uncovered what may be a need to regulate also on "changes to a major feature or component."

BY MR. PARLER:

Q As far as you are aware, Mr. Angelo, to the best of your recollection, has this report of this task group on post-CP applications amendments, ever been implemented?

A No, I am not aware that it was ever implemented.

Q Did you have the occasion during your brief involvement in early 1977 on this task group to examine some of the earlier efforts in that area, the area of proposed changes to deal with the meaning of principal architectural and engineering criteria?

A Yes, I examined the earlier documents.

MR. PARLER: I'd like to mark for identification some of these earlier documents, which I have previously handed to you.

I will mark for identification as Exhibit 1073 a proposed Federal Register notice dated March 31st, 1970, commencing at page 5317 through 5318.

Exhibit 1073, among other things, refers to proposed amendments that were published on April the 16th, 1969, and it states that the proposed definition in Section 50.2 of the "principal architectural and engineering criteria was not included, and the reason that definition was not included," according to Exhibit 1073, is that it appears that the essential elements of the proposed design of the structures, systems and components of water-cooled nuclear power units, referred to in the proposed rule, that is in the proposed rule on April the 16th, 1969, would require further definition involving additional study.

[The document referred to was marked Exhibit 1073 for identification.]

MR. PARLER: Another document that I will mark for identification as Exhibit 1074 is a document from Karl Kneil, who was the chairman of a task force, and as the chairman he forwarded to Roger S. Boyd, who was then the Deputy Director, Division of Reactor Licensing, a task force report on Staff review of post-CP design changes.

[The document referred to was marked Exhibit 1074 for identification.]

MR. PARLER: This report was dated December the 23rd, 1975.

BY MR. PARLER:

Q I gather, Mr. Angelo, that the time that you worked on a task force of the same subject in 1977, that you were not aware that the earlier report that I just mentioned and marked for identification as Exhibit 1074, that that report had been implemented; is that right? Were you aware of the status of this -- here is what I am talking about.

[Handing document to witness.]

A My understanding is that this early work was not implemented.

Q All right.

Also I'd like to mark for identification as Exhibit 1075 a memorandum from Roger S. Boyd, who was then the Acting Director, Division of Reactor Licensing, to Ben C. Rusche, then the Director, Nuclear Reactor Regulation, subject, proposed method of handling post-CP design changes, dated January the 7th, 1976.

[The document referred to was marked Exhibit 1075 for identification.]

BY MR. PARLER:

Q This Exhibit 1075 forwarded to Mr. Rusche the task force report dated December the 23rd, 1975, which has been previously marked for identification as Exhibit 1074.

I'd like to mark for identification as Exhibit 1076 a memorandum from Mr. Rusche to Mr. Boyd, who at the time was the Director of the Division of Project Management, a memorandum, subject, post-CP design changes, dated June the 27th, 1976, which also deals with the December 23rd, 1975 task force report.

[The document referred to was marked Exhibit 1076 for identification.]

MR. PARLER: I'd like to mark for identification as Exhibit 1077 a memorandum or a letter from Mr. Robert D. Pollard to Mr. Felton, who was the Director of the Division of Rules and Records of the Nuclear Regulatory Commission, a Freedom of Information Act request, which refers, among other things, to a proposed change in the principal architectural and engineering criteria in one of the licensing proceedings under review.

[The document referred to was marked Exhibit 1077 for identification.]

MR. PARLER: Off the record just a second.

[Discussion off the record.]

MR. PARLER: I'd also like to mark for identification as Exhibit 1078 a memorandum from Mr. Joseph Gallo, who at the time was the Chief Hearing Counsel in the Office of the Executive Legal Director, to Mr. Richard C. DeYoung, who at that time was the Assistant Director for Light Water Reactors Group 1, Division of Reactor Licensing. The memorandum is entitled "Forked River Plant Modifications," and the date is November the 17th, 1975. And this memorandum also discusses some of the considerations, particularly the legal considerations involved, in the issue of what are the principal architectural and design criteria, and also some of the considerations involved in post-CP changes to an application.

BY MR. PARLER:

Q Now, as far as you are aware, Mr. Angelo, from your participation in the 1977 effort in the area that we have been talking about, again, as I probably have already asked you, are you aware that any of the recommendations of either task force, the '75 task force or the '77 task force, have been implemented?

A I'm not aware of any attempt to make a formal implementation of these recommendations.

Q Do you have any other comment to make on this issue or the issue of post-CP amendments which I would like, for the record, to indicate was not the issue that I discussed with you previously, which -- that is this issue of systems interaction, which is the primary reason for your deposition?

But with that understanding, do you have any comments on the basis of your recollection that you'd like to make about this other issue that I have -- these documents that I have just marked for identification deals with -- deal with? Apparently for some time now, well over a decade, there has been some efforts being made or some need perceived to have additional clarification in the area as to what is meant by principal architectural and engineering criteria.

Presumably the need has been manifested because of certain things, such as -- in connection with a determination that might have to be made as to what a construction permit means and if after a construction permit is issued, what requirements must be followed to amend the Preliminary Safety Analysis Report.

I've covered those things to kind of like refresh your recollection, but with that background which I have represented to you, do you have any general comments or observations that you would like to make about this subject?

A Yes. I believe the question of what does the construction permit represent appears to be vanishing as time goes on, in the sense that the Staff in their review of an application for a construction permit, really solicits by way of questions sufficient detail about the design to be convinced and assured that they know what the principal architectural and design features really are, as described in the application for construction permit.

The question really was, was all of that detail really principal engineering design, or was it -- some of it merely detail to illustrate and not really a principal feature. I think that was it.

But the application presents a lot more information than you could call principal features. Principal architect/ engineering design features.

Well, I, at this point, you know, after looking back over now these several years -- you wonder whether the problem is really the same problem any more. Maybe it's different.

Q Do you have anything else to add about any of the things that we have talked about here this morning?

A No.

Q Do you have any questions, Tom?

Off the record.

[Discussion off the record.]

MR. PARLER: On the record.

In conclusion, let me say that this is an ongoing investigation, and although I have completed the questions I have for you today, we may need to bring you back for further depositions.

We will, however, make every effort to avoid having to do so.

I will now recess this deposition, rather than terminate it, and I wish to thank you on behalf of the Special Inquiry Group for your time and your cooperation in being with us here today, Mr. Angelo. Thank you.

[Whereupon, at 12:50 p.m., the deposition was adjourned.]

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Exhibit 1067

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April 30, 1974

Working CT-373

Mr. William Kerr
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D.C. 20545

POOR ORIGINAL

Dear Bill:

This response to your letter of December 5, 1973, started as a set of expanded comments on the Regulatory Guide on Physical Independence of Electrical Systems (now R.G. 1.75). In re-reviewing this document, it occurred to me that this Guide in itself represented the product of a more or less isolated activity which did not adequately interface with other activities in a total effort commonly called "system engineering." I think the need for this kind of engineering can be recognized and some steps taken to improve nuclear plant designs in this sense. This is the main purpose of this letter.

Since we are now entering a standardization phase of nuclear plant design, it seems more important than ever to recognize the plants as entities with well-organized, technically coherent and "balanced" concepts of "nuclear safety."

The designs I have seen and worked so far on have the common deficiency that they were not designed nor reviewed with the techniques of system engineering or "system analysis." They were handled by design and design assessment teams broken into the old "homogeneous" or functional engineering specialties of "civil," "electrical," "mechanical," and similar separated disciplines. The efforts of these groups has been and still are coordinated by production managers who have the prime objectives of minimizing costs and schedule delays. It follows that conflicts in the design contributions of these groups have typically not been searched out since significant findings would only create problems in costs and schedules. If the designs passed the AEC scrutiny that was good enough.

As you well know, we are still going through convulsions as a result of too late application of systems engineering principles to Class III and IV failures "outside of containment." The designs of main steam and feedwater systems in PWR's have recently undergone drastic changes to upgrade them to requirements compatible with system responsibility as a result of some similar attention to system engineering.

In evaluating the various faulted conditions, in finding bases for fire protection and industrial security, and in such activities as increasing circuit reliability as implied by this guide, there is the common goal as set forth in the enclosed ANSI N18.8 (Enclosure 5) of "limiting the mobility of radioactive material."

The subordinate goals or operating requirements for controlling the degree of "mobility" of radioactive materials and the "minimizing the risk" of gross loss of control (not well covered in ANS 18.8) are more complicated to identify and define. The fire or security expert or, in this case the electrical engineer, has no feel for what he is preventing by his efforts and thus doesn't know how conservative he should be in his design. He certainly must feel that "others" will probably compromise the reliability of his part of the design and he may easily compromise more conservative designs of others.

The current emphasis on standard plants suggests that intensive "system engineering" which bridges the too-independent engineering disciplines can now be applied to fewer plants so that the designers and any other interested and responsible parties could better understand what they and others are doing. However, the shift to standard plants also suggests significant shifts in responsibility, interest, and capability to do system engineering toward AEC and away from other participants, especially the using utilities.

Even now, in the course of developing "standard" designs, the incentives are disappearing to produce designs any better than those offered by the NSS vendors and minimally acceptable to AEC. These designs do not use system engineering in the connotation and degree represented by the enclosed comments.

It seems reasonable to me that at least the new standard nuclear plant designs should represent a much higher level of system engineering than is apparent in past and current designs. These comments are intended to help the process.

POOR ORIGINAL

I have marked this letter and the comments for "Official Use Only" and am sending similarly marked copies to 1 since the comment scope extends beyond the work of the subcommittee into areas 1 and I have discussed earlier. I understand that Mr. Tom Ippolito will attend the closed session of the subcommittee on May 8, and I have enclosed an extra copy of the material to be forwarded to him or any other as you deem appropriate.

Very truly yours,

1

Enclosure: One copy of comments and marked Guide.
One copy of 5 numbered enclosures.

CC: Dr. David Okrent
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D.C. 20545

POOR ORIGINAL

J Angelo

RLW

RLW Exhibit 1068

COPY

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

January 17, 1979

This report may contain proprietary or other information which should be protected from public disclosure. It should not be released until a review of the document has been completed.

TO: M. Bender, Chairman
Plant Arrangements Subcommittee

FROM: C. Michelson RLW for

SUBJECT: REMARKS CONCERNING SUBCOMMITTEE MEETING ON PLANT ARRANGEMENTS, OCTOBER 25, 1978

POOR ORIGINAL

The following remarks are concerned with the presentations which were made at the subject subcommittee meeting. I hope they will add further perspective to the important problem of systems interaction and aid the subcommittee in their deliberations.

I. REMARKS CONCERNING PRESENTATION BY JACK HICKMAN AND WALLY CRAMOND ON THE SANDIA STUDY OF SYSTEMS INTERACTION

Sandia seems to approach the problem of systems interaction by identifying which systems are important to safety and then trying to determine the safety significant interactions that might occur involving these systems and evaluate their effect on the performance of required safety functions. This is a logical way to approach the problem if its pitfalls are recognized and appropriately accommodated. In order to limit the scope of work to a manageable size, the study is being narrowed to events during normal plant operations and off-normal incidents of moderate frequency. The apparent intention is to develop a methodology that can be applied later to other plant conditions. The objective seems to be to develop a broadly applicable methodology as quickly as possible and use it to verify that the NRC Standard Review Plan and industry methods of handling the systems interaction problem are adequate.

As a further clarification of scope, Sandia is apparently concentrating on systems interaction that might significantly reduce the ability to (1) shut down the reactor, (2) remove decay heat, or (3) protect the reactor coolant system and prevent a LOCA. These are certainly the priority safety performance objectives for any plant involved in a non-LOCA event irrespective of interaction effects.

I have no real concern with this basic approach to the problem, as opposed to alternative ways. It is clearly straightforward and, therefore, amendable to methodical development, but I can forsee certain difficulties and limitations such as discussed below:

1. Interactions are considered to arise from the existence of commonalities. Some of these commonalities are easy to see but others are deeply hidden in the design or arrangement and are difficult to identify until after they expose themselves. Unfortunately, the exposure is often associated with a safety-significant event. One commonality of concern is associated

X-Michelson

RH-1

with "environmental conductors." These are conductors which are capable of transferring adverse environmental influences from one area to another during an event. Such conductors may appear obvious such as for the case of the back flow of hot air, vapor, or combustible gas through a ventilation duct following a fan failure. In other cases, the conductor may be subtle such as when water flows through an electrical conduit to a remote electrical board following the failure of a water-cooled component. These environmental conductors need to be identified if the fault trees are to be complete.

2. Normal plant operations include maintenance and testing. Associated with these operations is the requirement to use off-normal system alignments and procedures such as prescribed by the plant technical specifications. These alignments and procedures need to be examined to determine if unique interaction possibilities are established. Since the number of maintenance and test possibilities and their various allowable combinations are large and plant specific, it is not clear how this could be factored into the study. However, because a significant fraction of total plant operating time will involve such operations, it is not clear that they can be disregarded.

3. If the Sandia study is to include incidents of moderate frequency, it would appear that some consideration needs to be given to system interaction possibilities associated with operator errors, such as malalignments, which are normally treated as plant upset conditions. Since the number of possibilities for such errors is limitless, it may be difficult to handle as a requirement; but it is not clear that it should be disregarded.

4. The failure mode of a component can have an important influence on the extent of interactions which might result during the event. For instance, severe ancing in electrical switchgear during failure would produce electromagnetic radiation which may interact adversely with solid states control modules in the area. The rupture of a water line associated with a component could produce a water spray on a number of adjacent components. The leakage of a hydrogen cover gas line could lead to a flash fire or explosion with extensive interactions. Such effects are difficult to predict and account for in a study, but appear to be associated with normal plant operation. It is not clear how such spacial couplings will be handled.

5. The importance or effect of spacial coupling should be determined, in part, by the susceptability of the components involved to whatever the challenge might be in the space. For instance, a spacial coupling based on the development of a water spray during component failure cannot exist if all potential targets are resistant to spray. Similar arguments pertain to other spacial challenges such as elevated temperature, flooding, electromagnetic radiation, and steam releases. Unfortunately, detailed information concerning such environmental effects on components is often lacking.

6. It is not clear as to what extent and how possible single active component failures will be included when looking for possible interactions following a given initiating event within the scope of study. Such single failures are generally included in the plant safety analysis and might involve spacial or physical couplings through otherwise unrelated systems. How will these interactions be handled?

7. Interactions may result from degradations in the quality characteristics of essential supporting auxiliary services such as electric power, cooling water, and control air. The consideration here is not a loss of these services but their degradation. Large variations in voltage, frequency, water pressure, etc. can adversely affect the performance characteristics of components and systems and introduce interactions which can affect many systems. It is not clear how the Sandia study will uncover such interaction possibilities.

8. The total loss of essential supporting auxiliary services such as electric power, cooling water, and control air is also an important consideration, but, in some cases, the interruption effects are more dramatic if only a selective loss is incurred. For example, if electric power is lost to control instruments but not to control logic, the logic will attempt to respond to the failure mode of the instrument (upscale, downscale, or as is) and produce unusual control responses. Misinformation may also be supplied to the operator; it is not clear that the Sandia work will expose interactions of this kind.

9. Essential supporting auxiliary services are also subject to interaction effects resulting from automatic transfers, load shedding, or load additions. Such maneuvers have a potential to overload essential services through failure to isolate or the addition of unwanted loads. Some of the load shedding in cooling water and control air systems may involve non-qualified loads. In many cases, the normal supply for the service is non-

9. qualified with some type of automatic transfer to the qualified source. The ultimate effect of overloading might be a degradation in the quality characteristics of the auxiliary service or a partial or total loss of the service. Any one of these effects may produce adverse interactions which need to be included in the study. It is not clear that they already appear in the fault trees. Interactions between the auxiliary supply services should also be considered.

10. It is not clear that the Sandia methodology will take account of the cause and effect relationships which may develop sequentially as a result of interactions occurring during an event. If the interaction effect of one cause creates another cause and interaction effect, etc., then the interactions should be evaluated in proper sequence. This certainly complicates the fault tree and its programming.

11. An accidental actuation of systems such as fire protecti or containment spray should be treated as a plant upset condition and evaluated for possible adverse interaction effects. It is not clear that this will be included in the intended scope of study.

12. Reactor vessel head removal and refueling are modes of normal operation and should receive attention relative to possible adverse interactions. Head removal is of special concern since it represents a duration of jeopardy during which pressurization is not possible (e.g. head bolts may be loose) and the steam generators are no longer functional as heat sinks. The reactor core cannot communicate effectively with the steam generators by convective flow or an evaporation/condensation process. The core decay heat must be removed by the decay heat removal system. A failure to do so could lead to dangerous modes of heat removal. The unique nature of the plant alignment, physical configuration, and operational procedures during this time could give rise to unusual interaction possibilities that might otherwise escape notice. During and after head removal, the interaction studies should also extend to systems involved in the handling of heavy objects which could be dropped into the open core.

13. Apparently human interactions with systems will be included in the Sandia fault trees, but I can foresee a real complication in modeling man/machine interaction situations such as operator response based on misinformation, or operator reacting to conf cting information.

14. An area of unusual interaction complication is the interface between the process systems and their control and protection systems. Involved here are interfaces with both qualified and non-qualified controls and with the human operator. Some of the networks involved are very complex and would be difficult to adequately model in a fault tree. Many of the networks must be treated as "black boxes" for manageable simplicity. The widespread use of solid states control modules further complicates the problem because their spacial interfaces are susceptible to environmental changes. Also involved is the plant computer and the plant solid states control system with its many human and process interfaces and multiple opportunities for spacial and physical interaction. Of particular concern is its potential vulnerability and fast adverse response to human error during on-line maintenance (e.g., dropping an indicating light bulb). It is not clear how and to what extent Sandia will include such items in their fault trees.

II. GENERAL REMARKS CONCERNING NRC PROGRAM FOR IDENTIFYING SYSTEM INTERACTIONS

This whole question of system interactions is rather complex and the possible breadth of consideration could be virtually limitless. Fault trees could be developed to include almost any concern, but, somehow, the scope of such a study must be confined to reasonable limits. My various comments concerning the Sandia work are intended to help identify the potential scope of this problem and thereby exemplify the shortcomings involved when striving for simplicity. They should not be interpreted as a recommendation for an expanded scope and they are not intended to detract from the high quality of the work being done. The work being performed by Sandia appears to be developing along rigorous academic lines with well defined bounds based on resource limitations and NRC safety priorities. However, the methodological procedures being developed do not appear promising at this time as practical tools for a plant designer or reviewer. They already seem rather complex to use and probably have limitations which, if overcome, would only add to their complexity.

I can forsee the Sandia methodology as a useful means for an in-depth study of adverse interactions on a limited scale. However, the work likely to be required in developing the unique features of the fault trees for an entire specific plant will probably make it a prohibitive technique for routine review purposes. Certain plants may share some common fault tree branches, but a large number will be plant specific and will most likely require considerable work to assure reliable evaluation results. Of course, it is still important to find out how far such a methodology can be developed and applied to produce realistic and useful results.

The system interactions of particular concern during plant design and safety review are those which are difficult to predict and find by simple inspection and which are safety significant. For these, appropriate experience is one of the best tools available to the reviewer. The problem is, however, that it is usually difficult to acquire the appropriate experience. This might be done through a prolonged exposure to the nuclear plant design review process with special emphasis on developing an in-depth understanding of how the plant responds to various postulated events and how the safety systems function for each case. If during this exposure various adverse plant interactions are uncovered and resolved, the experience acquired thereby will tend to sharpen the reviewer's ability to uncover additional, but similar, interactions and eventaully develop a higher degree of sophistication and sensitivity to the more subtle interaction possibilities. This is likely to require a prolonged work exposure and may not provide the needed experience unless the mission of the reviewer is to seek out such interactions and he is provided with dedicated supervision and resources with which to do the job.

While striving to acquire appropriate experience, there are some important assists available to help expedite the process. For instance, Licensee Event Reports can provide valuable insight into the kinds of interactions which might occur and thereby aid the reviewer in uncovering similar possibilities in other plants and enhance the learning process. Experiences acquired during plant preoperational and startup testing can prove invaluable in developing the needed depth of understanding of how the plant and safety systems behave and, to some extent, help to uncover interaction possibilities. Academic studies and tools such as provided by Sandia may also prove helpful in the training process by providing a theoretical basis for how some of the interactions come about and yield additional examples of what to look for. However, in my opinion it takes a proper mix of these various activities to develop the appropriate experience needed by the designer and reviewer to assure an adequate treatment of the systems interaction problem. The methodological techniques such as being developed by Sandia should not be considered as the principal tool.

Having acquired the appropriate experience, it is essential to conccentrate it in a dedicated organizational unit whose mission is clearly systems interaction oriented. It is from such an experienced unit that we could expect the development of better and more practical methods for handling the interaction problem. These methods might include additional analytical techniques, but a more promising output might be the publication of system interaction case studies based on actual experience and exemplifying the kinds of interaction problems that have been uncovered and how they are handled. Such case studies could be distributed like the "Operating Experience Bulletins" and would help to develop a competent experience base throughout the industry. The costs involved in pulling together the appropriate experience into an adequately staffed unit will probably exceed the reasonable expectations of most utilities. It appears that the NRC is in the best position to provide the needed continuing effort (either in-house or under contract).

III. REMARKS CONCERNING PRESENTATION BY JOHN ANDERSON ON ORNL WORK

ORNL seems to approach the problem of control system/protection system interactions by looking for the direct interactions between these systems and not the subtle ones. They are looking at failures and degradations, and evaluating their effects. They are not using formalized fault trees. As an alternative to the Sandia work, this is also a logical way to approach the interaction problem and should provide useful results. It represents another input to the activity mix needed to develop an appropriate experience base. It may, however, be limited in its depth of consideration. I have no specific comments on the ORNL work.

IV. REMARKS CONCERNING PRESENTATIONS BY JERRY VELLENDER AND CORDELL REED ON ZION SYSTEM INTERACTIONS STUDY

The Commonwealth of Edison Co. Zion System Interactions Study was based on a review of over 9,000 Licensee Event Reports of which 267 were considered to be applicable to Zion and 67 of which were selected for detailed consideration by Fluor Pioneer, Inc. The study concentrated on interactions relating to failures that could interfere with shutdown heat removal. The technique was to look at each LER and determine if it had impaired or degraded non-accident heat removal. If so, it was determined if it could happen at Zion and what corrective action might be needed. I would like to make the following observations and comments concerning this study.

1. Although the work performed by Fluor Pioneer could be considered an independent review of the 67 LER situations selected by Commonwealth Edison, it should not be considered an independent review of the systems interaction potential at Zion. This could only be claimed if Fluor Pioneer had performed the data reduction on the 9,000 LER's and selected the appropriate ones for detailed consideration.

2. The data reduction was based on looking for those LER's which produced system interactions considered adverse to shutdown heat removal. It is my understanding that if no adverse interaction occurred, the LER was not selected for detailed study. This might be a reasonable decision where the equipment and plant arrangement are sufficiently similar to those at Zion. It is not reasonable or correct if certain differences should exist. For instance, if the LER under study is related to a flooding event for which the equipment involved is already designed to accommodate, no adverse interactions should result at that plant and the LER would not be selected for additional study. If, however, the comparable equipment at Zion is not designed for flooding, then an adverse interaction might be experienced and the LER should be selected for further study to make this determination. Other types of potential interactions are also sensitive to equipment design and layout differences which need to be considered. It is not clear how many of the nearly 9,000 LER's reviewed and discarded might be included in this category and should have been selected for further study. Unless suitably clarified, it should be considered a basic shortcoming of the study.

3. An examination of LER's amounts to an examination of the historical record. The corrective actions taken should assure that history will not repeat itself, but it does not assure freedom from other adverse systems interactions. Some of the most serious interactions may not have taken place yet at some plant, or there may be interaction sit-

uations unique to Zion that remain to be exposed. The Zion study is certainly well done and useful as a contribution to the needed case studies, but it should be recognized as very limited in scope if the desired objective is to uncover the full spectrum of potential adverse interactions at the plant before they become self evident.

4. It was pointed out as a major conclusion in the Fluor Pioneer report that generic studies such as requested by the NRC for pipe breaks had already resulted in modifications to Zion which substantially reduced adverse interactions to such events. If this is the case, then I am somewhat surprised that the systems interaction study uncovered a problem with entry and accumulation of water in electrical boxes. I would have thought that water released as sprays, jets, or cascades during the pipe break studies would very likely enter some of these boxes and flag the drain hole problem for corrective action.

5. Since the control air system at Zion was not included in the list of systems for consideration, I assume it is classified as non-essential. If so, it is an important example of a non-safety system which may have a potential for safety-related system interactions which should be evaluated. PWR's of the Zion class usually make widespread use of air operated valves for process isolation and control for both the NSSS supply (Westinghouse) and the BOP (AE design). On loss of control air, these valves revert to safe positions as determined by an appropriate analysis. Such reversions may introduce safety-significant effects when one considers the number of valves and other control components undergoing simultaneous change, and the multiple loss of process control due to the control air failure. In some cases, both trains of redundant equipment may be involved and more than one unit in the plant. The acceptability of this loss must be evaluated using plant specific information and certain assumptions concerning manual operation. Special attention should be given to the effect on such important essential functions as auxiliary feedwater control, RCS chemical volume and control (makeup and letdown), and the continuation of acceptable performance for environmental control systems which are predominantly air operated (for dampers and process control). The loss of environmental control may interact adversely with such items as instrumentation and control (particularly where solid states modules are used), and electric power system components (e.g., motors, transformers, and switchgear control). In my opinion, the control air system should have been included for consideration even if classified as non-essential.

6. Other non-essential systems of concern and related to systems interaction are the non-IE electrical power (AC and DC), instrumentation and control, and plant computer systems. Although none of these systems are considered essential, they do interface strongly with the plant operator. Certain initiating events in these systems during normal operations can lead to extensive displays of maloperation, misinformation, and unwanted responses which must be interpreted and corrected quickly by the plant operator. If left uncorrected, they may lead to safety-significant degraded conditions. It is not clear that such non-essential systems were included in the interaction study. Although perhaps less critical than the control air system, I believe they should have been considered.

V. GENERAL REMARKS CONCERNING NRC PROGRAM FOR REVIEW OF LICENSEE EVENT REPORTS

The NRC program for review of LER's is an important aspect of the systems interaction work, but its present scope within NRC is unclear. Perhaps the subcommittee may wish to ask for a short presentation on this subject at a future meeting. For now, I would like to make the following observations for your consideration.

1. The LER's are an important source of real world information which should receive careful evaluation from the viewpoint of uncovering possible systems interactions and providing a feedback of information to the designers and reviewers. The Zion Interaction Study is an example of how this information might be evaluated and used for corrective actions. Eventhough it is "after the fact" information, it is still useful. Additional and more comprehensive work of this type needs to be done if the nuclear industry is to benefit fully from this past experience. In my opinion, the NRC is in the best position to have these studies performed (either in-house or under contract). They have the resources and recognized access to all information and facilities, and are in the best position to monitor the entire industry and thereby predict generic difficulties. According to the NRC people, some work on the LER's is being done within NRC, but is does not appear to me that it is adequately dedicated to a determination of possible adverse systems interaction.

2. Perhaps the problem of evaluating LER's for systems interaction could be somewhat eased if the preparer of the LER were required to indicate whether or not a system interaction was involved in the event before giving the details. This should not add significantly to the work of preparing the report and it would make the sorting a lot easier. The main problem is assuring that the preparer of the LER understands the concept of "systems interaction." It would be necessary for the NRC to define the concept with suffiicent clarity to assure consistent usage. This may not be easy, but progress is being made and the concept should become clearer as the principles and examples are developed.

3. As the situation now stands, it appears that the nuclear industry does not have or intend to have an organized effort to review and evaluate LER's for possible systems interaction as was done in the Zion study. The NRC is reviewing each LER but the scope of this review is not clear. This may mean that valuable experiences are not being adequately utilized from the viewpoint of the systems interaction program. Perhaps the subcommittee could benefit from presentations by the industry on how they view the problem and what they would propose to be done.

Exhibit 1085

JOHN ANGELO

PROFESSIONAL QUALIFICATIONS

LIGHT WATER REACTORS BRANCH NO. 1

DIVISION OF PROJECT MANAGEMENT

I am a Senior Project Manager in Light Water Reactors Branch No. 1 of the Division of Project Management, U. S. Nuclear Regulatory Commission. I am responsible for the evaluation of nuclear safety aspects of nuclear reactor facilities and serve as the project manager for technical evaluation of nuclear power reactor license applications.

I was born in Old Bethany, Pennsylvania. I am a graduate of the University of Idaho and received a Bachelor of Science in Electrical Engineering in 1949. I have done graduate engineering and science study at Rensselaer Polytechnic Institute, Union College and George Washington University and received a Master of Science in Engineering from Union College in 1963.

From 1949 to 1958 I was employed in the Large Steam Turbine-Generator Division and the Apparatus Sales Division of the General Electric Company progressively as a Performance Test Engineer, Turbine Supervisor, and Development Engineer. From 1960 to 1962 I was employed as a mechanical engineer with the Nuclear Power Division of ALCO Products, Inc. with responsibilities for the design and analyses of hydraulic, thermal and mechanical systems and components of nuclear power reactors. I continued these duties in 1962 with the Nuclear Power Division of Allis-Chalmers Manufacturing Company when that company purchased the nuclear business from ALCO Products.

In 1963 and 1964 I was employed as a mechanical engineer with the U. S. Army Engineer Reactors Group at Fort Belvoir, Virginia with broad responsibilities for design, operations, maintenance and safety reviews of nuclear power reactors for military applications.

I transferred to the Naval Nuclear Power Unit at Fort Belvoir, Virginia in 1964 as Director of the Technical Support Department from 1964 to 1967. From 1967 to 1972 I was Branch Chief of the Nuclear Engineering Branch of the Naval Facilities Engineering Command in Arlington, Virginia. In these two positions with the Department of the Navy I had supervisory responsibilities in nuclear, mechanical, electrical, and chemical engineering as applied to the design, development, operations, maintenance and modifications of nuclear power reactors and radioisotope power generators. I had direct responsibility for nuclear safety analysis and reviews.

In January 1972, I accepted an appointment as Project Manager with the Directorate of Licensing and have been assigned responsibilities for the safety review of Beaver Valley Power Station, Byron Station, Braidwood Station, Fluor Pioneer Standard Balance of Plant, and have assisted in the project management of several other projects. In June 1977 I was assigned as Task Manager for Generic Task A-17, Systems Interaction in Nuclear Power Plants, with responsibility for the development of the task and its technical accomplishment.

I am a registered professional engineer in the State of Massachusetts.

J. Angelo
238

Exhibit 1066

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

JUN 28 1977

E. G. Case, Acting Director
Office of Nuclear Reactor Regulation

CORRESPONDENCE REGARDING SYSTEMS INTERACTION ANALYSIS

Attached in accordance with our discussion is a package of correspondence related to the analysis of systems interactions that have been recommended by the Committee.

R. F. Fraley
R. F. Fraley
Executive Director

Attachments:
1. Ltr. to L. V. Gossick frm. M. Bender dtd. 6/17/77 re. Review of Systems Interaction
2. Memo to E. G. Case frm. R. Fraley dtd. 5/12/77 re. Reliability of Power Supplies
3. Memo to E. G. Case frm. R. F. Fraley dtd. 4/26/77 re. DC System Reliability
4. Ltr. to L. V. Gossick frm. M. Bender dtd. 3/15/77 re. Reliability of Power Supplies
5. Ltr. to L. V. Gossick frm. M. Bender dtd. 3/15/77 re. Auxiliary System Reliability
6. Memo to L. V. Gossick frm. R. F. Fraley dtd. 1/19/77 re. Topics for Discussion During ACRS-NRC Meeting
7. Ltr. to M. A. Rowden frm. D. W. Moeller, dtd. 12/17/76 re. GESSAR-238 and GESSAR-251
8. Memo to L. V. Gossick frm. R. F. Fraley, dtd. 11/1/76 re. Analysis of Systems Interactions

(Continued on next page)

9 Ltr. to M. A. Rowden frm. D. W. Moeller, dtd. 10/22/76 re. Three Mile Island, Unit 2
10. Ltr. to M. A. Rowden frm. D. W. Moeller, dtd. 7/14/76 re. RESAR-3S
11. Ltr. to M. A. Rowden frm. D. W. Moeller, dtd. 6/11/76 re. SWESSAR-P1
12. Ltr. to W. A. Anders frm. D. W. Moeller, dtd. 2/11/76 re. SWESSAR-P1
13. Ltr. to W. A. Anders frm. W. Kerr, dtd. 9/17/75 CESSAR-80
14. Ltr. to W. A. Anders frm. W. Kerr, dtd. 3/14/76 re. GESSAR-238
15. Ltr. to L. M. Muntzing frm. W. R. Stratton, dtd. 11/8/74 re. Systems Analysis of Engineered Safety Systems

cc: w/atts:
L. Crocker, DPM
R. Heineman, DSS

cc: w/o atts:
M. Bender, ACRS
J. Ebersole, ACRS
S. Lawroski, ACRS

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

June 17, 1977

Mr. Lee V. Gossick
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REVIEW OF SYSTEMS INTERACTION

Dear Mr. Gossick:

During the recent review of the operations of the Zion Station, Units 1 and 2, a question was raised regarding the scope of the systems interaction review recommended by the Committee in its report of June 9, 1976.

In order to provide clarification of its intent with regard to systems interaction evaluation, the ACRS offers the following two examples of possible studies:

1. Examine the physical configuration of safety systems (a) in relation to their presumed "redundant" divisions or channels, (b) in relation to their supportive sub-functions, and (c) in relation to non-safety systems and features, for actions or consequences in one that have a direct or indirect deleterious effect on another. Such configurations might allow failures or local hostile conditions to unduly interfere with the minimum functions required to remove decay heat after shutdown. Particular attention should be given to the potential for "cascading" failures leading to a terminal event which interferes with some aspect of the shutdown functions.

2. Examine interrelated functions and actions as they relate to operating practices, such as the recent action which caused the burn-out of a diesel-generator during a loading test. This would include reevaluation of Technical Specifications to ascertain whether undue degradation of minimum shutdown heat removal capability may be occurring because of unrecognized indirect connections between systems. Another example would be the mechanical maintenance of a valve or pump in one train

concurrent with electrical maintenance of switchboards or relay panels in the "redundant" train which is inactive but supposed to be immediately available.

Sincerely yours,

M. Bender

M. Bender
Chairman

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

May 12, 1977

E. G. Case, Acting Director
Office of Nuclear Reactor Regulation

RELIABILITY OF POWER SUPPLIES

References:
(1) Memo from M. Bender, ACRS Chairman to L. V. Gossick, EDO, "Reliability of Power Supplies," dtd. March 15, 1977
(2) Memo from R. F. Fraley, ACRS, to E. G. Case, NRR, "D.C. System Reliability," dtd. April 26, 1977

The attached report from Mr. E. P. Epler, ACRS consultant, provides information applicable to the reevaluation of D.C. power supplies requested in References (1) and (2) above.

R. F. Fraley
Executive Director

Attachment:
Letter from E. P. Epler to J. C. Ebersole & D. Okrent dtd. April 12, 1977, re. D.C. power supply

cc:
R. Heineman, w/att.
R. Boyd, w/att.

ELBERT P. EPLER
NUCLEAR SYSTEMS CONSULTANT
712 FLORIDA AVENUE
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483-0994

RECEIVED

April 12 1977

J.C. Ebersole
D. Okrent

POOR ORIGINAL

The reliability of the D.C. power supply came under discussion at the April 8 meeting of the ACRS. The single failure criterion has been applied in the design of the battery system and it is the Regulatory position that reliability, so obtained, is adequate. It was suggested however that a reliability analysis should be performed. The following are my comments.

It is generally agreed that the availability of D.C. is essential to residual heat removal. It is important that the degree of this dependence be established; if it is indeed true that the loss of both batteries would invariably, or with high probability, lead to core melt, then this degree of dependence should be firmly and precisely established. It would then be required that the D.C. supply, which would be called upon for residual heat removal ten times per year, have an availability at least equivalent to that of the Reactor Shutdown System which is challenged no more than three times per year.

It has been observed that in most instances, when a single battery had failed, a scram resulted from the failure. It would seem therefore, that should both batteries fail, a scram would invariably follow. If indeed it is true that residual heat could not be removed on loss of both batteries, then a scram resulting from loss of both batteries would clearly be intolerable. It is therefore <u>mandatory</u> that loss of heat removal capability and reactor scram be made to be independent events.

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The failure of both batteries is not highly improbable. During the period Aug 31 1974 to Nov 3, 1976, approximately two years, there were 50 reportable occurrances relative to the D.C. supply. The following battery failures occurred.

<u>Date Reported</u> <u>Location</u>. <u>Event</u>

Nov 17 1976 Dresden 3 Battery found to be in degraded condition during refueling outage, caused by prior charger troubles. Battery replaced.

Nov 12 1976 Vt Yankee Battery voltage low - charger failure. Reactor at power, spare battery immediately connected.

Nov 3 1976 Palisades Bus voltage dropped to 60V. (parasitic) oil lift pump caused charger breaker to trip. Reactor in shutdown condition.

July 26 1976 Browns Ferry-2. Cell post broken off during cleaning. Reactor in shutdown condition however scram signal was generated.

July 26 1976 Robinson 2 Leads removed from batt. with reactor critical.

Sept 29 1976 Zion 2 With diesel generator in operation the loss of one DC bus prevented circuit breakers from removing excessive A.C. loads. As a result the diesel generator windings were destroye. Scram resulted.

Dec 23 1975 Oyster Creek. During a test on the station battery, the 125 V D.C. distribution center was deenergized with reactor at power. Shutdown resulted

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Aug 11 1975 Oconee 2 Simultaneous outage of two batteries. One string of Keowee and one string of switching station batteries were being charged simultaneously.

May 3 1975 Dresden 3 Discharge test during refueli found several bad cells in both batteries.

Oct 23 1974 Turkey Point 4 Discharge test found bad cells in one battery, caused by overcharging.

Sept 10 1974 Quad Cities (Parasitic) oil pump caused charger breaker to trip. Discharged battery caused scram.

Eleven failures occurred in approximately two years which works out to be 10^{-1} per reactor year. These were single battery failures, however two events could be categorized as incipient 2-battery failures.

In addition to the battery failures there were, during this period, 15 cases of charger failure which would lead to battery failure. Some mechanisms made their apperance which would affect both batteries.

1. The CO_2 Fire Protection System, when tested, caused thermal shock, causing battery jars to crack and leak electrolyte.

2. Surge voltage supressors failed and, more than once, caused failure of both battery chargers.

3. A charger was found to be operating at full capacity because of multiple leakage paths to ground through seismic braces

4. The handy bus tie breaker which connects the two batteries and thereby violates independence of the two D.C. supplies.

If the failure of one of two batteries would not cause reactor shutdown, but would be detected and the battery returned to service in ten hours, then the failure rate of 10^{-1} per year would yield an unavailability of 10^{-4} per year. This would make even a single battery almost acceptable. The independent failure of both batteries would then yield an unavailability of 10^{-8} per year. which would be entirely satisfactory. As we have seen however, the failure of one battery usually causes reactor shutdown and residual heat must be removed using the single remain. battery, and amid general confusion resulting from multiple equipment failures. Further, several mechanism have already made an appearance which would likely cause the failure of both batteries.

Summary

If it is confirmed that the D.C. supply is essential to residual heat removal, then the following principles applicable to Reactor Shutdown Systems would be applicable to the D.C. Supply.

1. Principle Separation of safety and control.
Remedy Remove from the batteries, dedicated to residual heat removal, all features which could cause, or lead to, reactor scram.

2 Principle Protection systems shall be dedicated and used for no purpose other than protection.
Remedy Remove all parasitic loads.

3 Principle Redundant channels, or trains, shall be independent.
Remedy Remove the bus tie breaker which invites the failure of both batteries.

4 Principle The system shall revert to a safe condition on loss of power.
Remedy No remedy is possible. As a minimum the charger should be sized so as to minimize the need for the now excessive protection, which is the principle cause of charger unavailability.

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Having made these corrections it would be appropriate to ask whether the D.C. system availability would be comparable to that of the Reactor Shutdown System which is itself deemed to, alone, be inadequate.

It is clear that the single failure criterion is inadequate to assure a reliable D.C. supply. It is also clear that a reliability analysis of the present system would yield no useful additional information and would result in unnecessary delay.

R P Effler

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

April 26, 1977

Edson G. Case Acting Director
Office of Nuclear Reactor Regulation

D.C. SYSTEM RELIABILITY

During the 204th ACRS meeting members of the NRC Staff reported to the Committee regarding the reevaluation of the battery supplied D.C. power system for Three Mile Island Nuclear Station, Unit 2 in response to the ACRS Report of October 22, 1976. Several Committee members expressed concern (Transcript pp 320-p333) with the basis for the Staff position and suggested that this reevaluation be conducted in accordance with the letter from Mr. M. Bender, Chairman, ACRS, to Mr. Lee V. Gossick, EDO, NRC, dated March 15, 1977, subject: Reliability of Power Supplies.

R. F. Fraley
R. F. Fraley
Executive Director

cc: L. Crocker, DPM
R. Boyd, DPM
R. DeYoung, DPM
R. E. Heineman, DSS

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

March 15, 1977

Mr. Lee V. Gossick
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: RELIABILITY OF POWER SUPPLIES

Dear Mr. Gossick:

The ACRS recommends that the NRC Staff review and evaluate the probability of loss of all AC power as a function of the duration of such power loss, and develop criteria and a specific approach to examining the capability of nuclear power plants to withstand such loss, as part of the standard NRC review for construction permits and operating licenses.

The estimates of probability of loss of AC power in WASH-1400 should be reviewed with a view toward determining whether AC power reliability can be estimated generically or should be evaluated on a case-by-case basis.

The Committee recommends also that the NRC Staff evaluate the reliability of typical minimum vital DC power arrangements and advise the Committee of the bases for judgments concerning their adequacy, and the possibilities for significant improvements in the reliability of such systems.

The ACRS is willing to work with the Staff via an appropriate Subcommittee in the evaluation and resolution of these matters.

Sincerely yours,

M. Bender

M. Bender
Chairman

POOR ORIGINAL

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

March 15, 1977

Mr. Lee V. Gossick
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: AUXILIARY SYSTEM RELIABILITY

Dear Mr. Gossick:

The ACRS believes that it is important to nuclear plant safety to understand better the reliability of those auxiliary systems necessary to establish and maintain reactors in the cold shutdown condition after loss of such primary full power heat sinks as steam generators and the main condenser.

Such auxiliary systems, which include Reactor Auxiliary Cooling Water Systems, Station Service Water Systems, and the Ultimate Heat Sink, generally are in continous duty in all modes of reactor operation and current practice allows these systems to be "redundant" two-train configurations. This results in dependence on a single system whenever one train of such a system fails or is shutdown for maintenance or other purposes.

As an example, to better understand the current situation, the ACRS requests that the NRC Staff perform an evaluation of the reliability of the typical minimum configuration allowed for the Station Service Water System and any alternate back-up systems, in order to quantify better the adequacy of design and to ascertain the potential for substantive improvements in overall safety by design changes for future plants.

Sincerely yours, POOR ORIGINAL

M. Bender

M. Bender
Chairman

January 19, 1977

Lee V. Gossick
Executive Director for Operations

TOPICS FOR DISCUSSION DURING ACRS-NRC MEETING

During the 201st ACRS meeting members of the Committee identified the following matters as items deserving further discussion with the NRC Staff. Time has tentatively been set aside during the 202nd Committee meeting to discuss these items to the degree that the NRC Staff is prepared to do so.

1. Study of the probability of steam generator tube failures in PWR plants and the consequences of such failures on the course of accidents such as LOCA-ECCS transients, and steam line break accidents.

2. Evaluation of the radiological consequences of steam line break accidents for the various types of PWR plants. A comparison should be made of the B&W once-through steam generator with similarly sized Westinghouse and Combustion Engineering units. This evaluation should include an analysis of the reduction in fission product removal resulting from the reduced inventory of water in a once-through steam generator and consideration of any specific design features or operating limits which may be appropriate to compensate for this reduced decontamination factor.

3. Study of the probability and consequences of the loss of feedwater control and resultant flooding of the superheater section of B&W once-through steam generators. Consideration should be given to: (a) the effects on core reactivity resulting from a rapid reduction in primary system temperature by the rapid increase in primary to secondary heat transfer; (b) the effects on steam driven equipment such as main and auxiliary feed pumps and the dynamic effects on system piping by the sudden introduction of water slugs into the steam system; and (c) the potential for and consequences of secondary system overpressure resulting from a turbine trip or loss of offsite power resulting from the transient caused by this rapid increase in steam generating capacity.

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OFFICE ▶						
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4. Evaluation of the thermal stresses imposed on the reactor pressure vessel by a steam line break aggravated by flooding of the superheater section of the B&W once-through steam generator. An analysis should be made of the decrease in the primary coolant inlet temperature resulting from such a transient and the effect it would have on stresses in the pressure vessel.

   In the event flooding of the steam generator is considered an independent event, verification should be provided that the feedwater control system will function properly during a steam line break transient.

5. Evaluation and application of Federal Republic of Germany (FRG) reactor safety research program results and FRG regulatory requirements to the NRC regulatory process. The FRG has an extensive safety research effort in progress and is rapidly developing an extensive body of information, criteria and regulatory requirements which are applicable to light water cooled reactors of the type being licensed in the U.S. Consideration should be given to a systematic program for exchange of information and consideration of this information as it applies to the NRC regulatory program.

6. Evaluation of fuel handling accidents inside containment. During the review of the Operating License for the North Anna Power Station Units 1 and 2, a question was raised regarding evaluation of a fuel handling accident inside containment. The Committee concluded that this issue could be resolved on a generic basis prior to the first refueling at this station. It is our understanding that a similar question has been raised of the Commission by the Union of Concerned Scientistsists and has been referred to the NRC Staff for evaluation on a generic basis. The Committee is interested in a report on the results of this evaluation or, if it is incomplete, the schedule for its completion.

Please let me know to what degree the Staff will be prepared to discuss these items during the 202nd ACRS meeting so appropriate arrangements can be made.

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R. F. Fraley
Executive Director

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

December 17, 1976

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

Subject: GENERAL ELECTRIC STANDARD NUCLEAR STEAM SUPPLY SYSTEMS
(GESSAR-238 NSSS and GESSAR-251)

POOR ORIGINAL

Dear Mr. Rowden:

At its 200th meeting, December 9-11, 1976, the Advisory Committee on Reactor Safeguards completed its review of the applications by the General Electric Company for preliminary design approvals for the standardized nuclear steam supply systems described in the General Electric Standard Safety Analysis Reports, GESSAR-238 NSSS and GESSAR-251. These systems were also considered at the 199th meeting of the Committee, November 11-13, 1976. The review by the Committee was facilitated by the meeting of a Subcommittee with representatives of the General Electric Company and with members of the Nuclear Regulatory Commission (NRC) Staff on November 6 and 7, 1976, in Los Angeles, California. The Committee also had the benefit of the documents listed below.

These GESSAR systems consist of BWR/6 nuclear steam supply systems with thermal power ratings of 3579 MW for GESSAR-238 NSSS and 3800 MW for GESSAR-251. They do not include the containment structure, the reactor building, the fuel building, the auxiliary building, or other structures outside the nuclear systems, which are the responsibility of the utility-applicant.

GESSAR-238 NSSS and GESSAR-251 have essentially the same design, the principal differences arising from differences in power level. Many aspects of GESSAR-238 NSSS and GESSAR-251 are the same as those previously reviewed by the Committee in the General Electric Standard Safety Analysis Report for a nuclear island. The Committee's report on the GESSAR-238 Nuclear Island was issued on March 14, 1975.

The Committee and the Applicant discussed the possibility of damage to a heat exchanger of the Residual Heat Removal (RHR) system by overpressurization or by hydrodynamic forces that could conceivably result from valve malfunction or operator error. This hypothetical condition is associated with the steam condensing mode of operation of the RHR or when the Reactor Core Isolation Cooling system is in use. The Committee recommends that the NRC Staff review this problem, applying their usual criteria of assuming malfunction and operator error, to determine whether such conditions can occur. The review should include particular study of the initial phase of steam entry, during which water initially present, or inadvertently accumulated, must be expelled from piping into and through the heat exchanger.

The Committee recommends that GESSAR-238 NSSS and GESSAR-251 incorporate appropriate systems to mitigate the consequences of an ATWS event.

Safety related interfaces between the reference system and the balance of plant are identified in these GESSAR's. It will be necessary for the NRC Staff to assure that all of the safety-related requirements are fulfilled when an application for a construction permit is filed. The Committee will review this matter in more detail when applications for construction permits referencing these systems are received.

The Committee recommends that, during the design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for an entire spectrum of anticipated operations and postulated design basis accident conditions.

Other generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. Those problems relevant to large boiling water reactors should be dealt with appropriately by the NRC Staff and General Electric Company as solutions are found. The relevant items are: II-3, 4, 5, 6, 7, 8, 9, 10, 11; IIA-2, 6; IIB-2, 4; IIC-1, 2, 3, 4, 6, 7.

The Committee believes that, subject to the above comments and to successful completion of the necessary R&D programs, GESSAR-238 NSSS and GESSAR-251 can be successfully engineered to serve as reference systems.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

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References

1. 238 Nuclear Steam Supply System GESSAR and Amendments 1 through 4.

2. Report to the Advisory Committee on Reactor Safeguards in the Matter of General Electric Safety Analysis Report GESSAR-238 NSSS (Docket No. STN 50-550) Published: October 1976 by the U.S. Nuclear Regulatory Commission.

3. 251 General Electric Standard Safety Analysis Report (251 GESSAR) and Amendments 1 through 21.

4. Report to the Advisory Committee on Reactor Safeguards in the Matter of General Safety Analysis Report GESSAR-251 (Docket No. STN 50-531) Published: October 1976 by the U. S. Nuclear Regulatory Commission.

5. General Electric Company letter dated February 13, 1975 forwarding proprietary information regarding fuel assembly and core design.

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November 1, 1976

L. V. Gossick
Executive Director for Operations

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ANALYSIS OF SYSTEMS INTERACTIONS

The ACRS has established a subcommittee to look into the effects of systems interactions in nuclear power plants. Among the topics to be considered, at a future meeting, are:

1. Probability and consequences of multiple tube ruptures in a B&W once-through steam generator with concurrent loss of condenser function.

2. Operability of rotating machinery following a seismic event. This would include consideration of loads, deformations, etc., of shafts, bearings, seals, and nozzles (flanges) and any potential problems of starting a machine that was stationary during the shock loading.

3. Method of bringing a PWR from a high pressure condition to low pressure cooling assuming the use of only safety grade equipment.

4. The effect of PWR loop isolation valve closure during a LOCA on ECCS performance.

5. Ability of equipment and components in containment to perform their intended function when exposed to a LOCA environment after a period of aging and maintenance that involves the opening of sealed enclosures.

6. Consequences of the inadvertent release of hydrogen into the plant due to the failure of such things as the hydrogen charging lines for the main generator cooling or reactor coolant chemical control systems, etc.

7. Adequacy of provisions to provide physical separation of Component Cooling Water Systems which are vital to the performance of engineered safety system components.

AM 2.1

Please let me know of an estimated date when your Staff can be prepared to discuss the items noted above so that an appropriate meeting can be scheduled. John C. McKinley (Ext. 1371) will coordinate detailed arrangements for this meeting if your Staff requires additional information.

(S)

R. F. Fraley
Executive Director

cc: ACRS Members
S. Varga
L. Crocker
R. DeYoung
R. Boyd
R. Heineman
B. Rusche
E. Case
R. Minogue

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 22, 1976

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

SUBJECT: REPORT ON THREE MILE ISLAND NUCLEAR STATION, UNIT 2

Dear Mr. Rowden:

During its 198th meeting, October 14-16, 1976, the Advisory Committee on Reactor Safeguards completed its review of the application of the Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company (Applicants) for a license to operate Three Mile Island Nuclear Station, Unit 2. This project was also considered during a Subcommittee meeting held in Harrisburg, Pennsylvania, on September 23 and 24, 1976. Members of the Committee visited the facility on September 23, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, General Public Utilities Service Corporation, the Babcock and Wilcox Company (B&W), Burns and Rowe, Inc., and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had available the documents listed below. The Committee reported on the application for a construction permit for Unit 1 on January 17 and April 12, 1968, and for an operating license for Unit 1 on August 14, 1973. The Committee reported on the application for a construction permit for Unit 2 on July 17, 1969.

The Three Mile Island Nuclear Station, Units 1 and 2, is located on Three Mile Island near the eastern shore of the Susquehanna River, about 12 miles southeast of Harrisburg, Pennsylvania. About 2380 people live within a two-mile radius of the site (the low population zone). The minimum exclusion distance is 2000 feet. The nearest population center is Harrisburg (1970 population 68,000).

Several changes have been made to bring the Babcock and Wilcox Emergency Core Cooling System (ECCS) evaluation model into conformance with the requirements of 10 CFR 50.46, and Appendix K to Part 50. Analyses of a spectrum of break sizes appropriate to Three Mile Island, Unit 2 have been completed using the approved B&W generic evaluation model. The

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results of the analyses for the reactor coolant pump discharge break, believed to be the "worst" break, show maximum allowable linear heat generation rates as a function of elevation in the reactor core ranging from 15.5 to 18.0 kilowatts per foot. Corresponding calculated post-accident peak clad temperatures range from 2002°F to 2146°F. The NRC Staff has identified additional information that it will require to complete its review and the Applicants' submittal is expected by the end of 1976. The Applicants propose to use both in-core and ex-core instrumentation to assure accuracy of measurement of core power distributions. The Committee believes that the proposed monitoring methods may be acceptable, but that an augmented startup program should be employed, and that satisfactory experience at 100% steady state power and during transients at less than full power should be obtained. This experience should be reviewed and evaluated by the NRC Staff prior to operating at up to full power in a load following mode. The Committee wishes to be kept informed.

A question has arisen concerning asymmetric loads on the reactor vessel and its internal structures for certain postulated loss-of-coolant accidents in pressurized water reactors. The Staff has required the Applicants to supply further information in order to complete its assessment of this matter. This issue should be resolved in a manner satisfactory to the NRC Staff.

The question of whether Unit 2 requires design modifications in order to comply with WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors", remains an outstanding issue pending the NRC Staff's completion of its review of B&W generic analyses of anticipated transients without scram. The Committee recommends that the NRC Staff, the Applicants and B&W 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.

Emergency plans have been developed to allow plant shutdown and maintenance of safe shutdown in the event of a maximum probable flood. Such a postulated flood would top the levee surrounding the plant by several feet. Included in the plan is the fastening of water tight steel panels in doorways and other openings of safety related structures. The Committee believes that the details of this plan, particularly relating to re-entry into the station during the post-flood period, need to be more clearly delineated.

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The Committee supports the NRC Staff's program for evaluation of fire protection in accordance with Branch Technical Position APCSB 9.5-1, Appendix A, "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 Three Mile Island Unit 2 and 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.

The Committee notes that long-term post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptable 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 procedures to confirm continuous long-term seal capability should be developed.

The Committee recommends that further review be made of the battery supplied DC power system to assure that non-essential loads do not interfere with its safety function. The Committee recommends that further review be made to assure no unacceptable effects such as release of hydrogen into the plant can occur from the failure of a hydrogen charging line. The Committee also recommends that studies be made to assure that failure of an instrument line cannot cause plant controllability problems of significance to public safety.

The management organization proposed by the Applicants to delineate the safety related responsibilities of the off-site and on-site personnel of the Three Mile Island Station left open questions as to how these responsibilities are to be discharged during normal working hours and during evening, night, and weekend shifts. This matter should be resolved to the satisfaction of the NRC Staff.

The NRC Staff is still reviewing various issues related to accidents leading to loss of fluid in the steam generator secondary side, such as steam line breaks. The Committee wishes to be kept informed of the resolution of these issues.

The Committee recommends that, prior to commercial power operation of Three Mile Island Unit 2, additional means for evaluating the cause and likely course of various accidents, including those of very low

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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 Committee believes that the Applicants and the NRC Staff should further review the Three Mile Island Nuclear Station for measures that could significantly reduce the possibility and consequences of sabotage, and that such measures should be implemented where practical.

Other generic problems relating to large water reactors are discussed in the Committee's report entitled "Status of Generic Items Relating to Light Water Reactors: Report No. 4", dated April 16, 1976. Those problems relevant to the Three Mile Island Station should be dealt with appropriately by the NRC Staff and the Applicants as solutions are found. The relevant items are: II - 1, 2, 3, 4, 5, 6, 7, 9, 11; IIA - 1, 4, 5, 6, 7, 8; IIC - 1, 2, 3, 4, 5, 6, 7.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that Three Mile Island Nuclear Station, Unit 2 can be operated at power levels up to 2772 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

References

1. Three Mile Island Nuclear Station, Unit 2 Final Safety Analysis Report (April, 1974) with Amendments 1 through 44.

2. Safety Evaluation Report (NUREG-0107) related to operation of Three Mile Island Nuclear Station, Unit 2, dated September, 1976.

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

July 14, 1976

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

Subject: REPORT ON WESTINGHOUSE ELECTRIC CORPORATION REFERENCE SAFETY ANALYSIS REPORT, RESAR-3S

Dear Mr. Rowden:

At its 195th meeting, July 8-10, 1976, the Advisory Committee on Reactor Safeguards completed its review of the Westinghouse Electric Corporation's application for a Preliminary Design Approval (PDA) for a standardized nuclear steam supply system consisting of a pressurized water reactor as described in its Reference Safety Analysis Report, RESAR-3S. A subcommittee meeting was held with representatives of the Applicant and the Nuclear Regulatory Commission (NRC) Staff in Washington, DC, on June 16, 1976. The Committee had the benefit of discussions with representatives of the NRC Staff and the Westinghouse Electric Corporation. The Committee also had the benefit of the documents listed below.

RESAR-3S is a Westinghouse standardized four-loop, single-unit nuclear steam supply system with a core thermal power of 3411 MWt. Systems within the scope of RESAR-3S include the reactor core, reactor coolant system and supports, chemical and volume control system, emergency core cooling system, boron recycle system, residual heat removal system, fuel handling system, and associated instrumentation and controls for these systems. RESAR-3S is similar to the nuclear steam supply system of the SNUPPS projects, reviewed in ACRS reports of September 17, October 16, and December 11, 1975. The ACRS report of September 18, 1975 reviewed the Westinghouse nuclear steam supply system RESAR-41. Significant features, other than those associated with the higher power level, which were incorporated in RESAR-41 but are not in RESAR-3S, include longer fuel assemblies, a rapid refueling system, an emergency boration system, and the use of three independent injection trains in the emergency core cooling system (ECCS).

RESAR-3S has been designed for application to an envelope of plant sites which includes provision for a Safe Shutdown Earthquake with a maximum horizontal ground acceleration of 0.4g.

RESAR-3S provides for those safety-related interface requirements that are essential to designing the balance of plant to be consistent with the assumptions used in the accident analyses. Since the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related design requirements have been met, these matters will be reviewed in more detail with the utility-applicants on a case-by-case basis. The Committee recommends that during design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for the entire spectrum of anticipated operations and postulated design basis accident conditions. For multiple reactor units at a single station, the Committee anticipates that safety-related items in RESAR-3S would be separately provided for each reactor unit.

An issue to be resolved prior to preliminary design approval for RESAR-3S involves the possibility of a single failure leading to the loss of the residual heat removal system. The Committee recommends that this matter be resolved in a manner satisfactory to the NRC Staff and wishes to be kept informed.

The Committee recommends that Westinghouse emphasize analytical and experimental programs to substantiate the conservatisms in the current Westinghouse ECCS evaluation model and to establish the accuracy and uncertainties in their best-estimate calculations. Timely progress reports should be provided on the performance of the 17x17 fuel, the control systems, improvements in the best estimate analyses, test verification of analytical methods, and reliability studies undertaken to establish meaningful improvements in components, systems, and arrangements for ECC systems and the dependent auxiliaries necessary to sustain the heat transport systems. The Committee recommends that if studies establish that ECCS improvements, such as obtainable from higher reflooding rates, can be achieved, consideration should be given to incorporating them into RESAR-3S.

Further review should be made on the adequacy of the RESAR-3S provisions for the maintenance, inspection, and operational needs of the plant throughout its service life and for eventual decommissioning. In particular, the Committee believes that the NRC Staff and the Westinghouse Electric Corporation should review methods and procedures for minimizing, and, if necessary, for removing accumulations of radioactive contamination so that maintenance and inspection programs can be more effectively and safely carried out.

The Committee believes that Westinghouse and the NRC Staff should continue to review RESAR-3S for design changes that will further improve protection against sabotage.

Generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. The Committee believes that procedures should be developed to incorporate approved resolution of these items into RESAR-3S.

The Committee believes that, subject to the above comments, RESAR-3S can be successfully engineered to serve as a reference system.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES

1. Westinghouse Electric Corporation, "Reference Safety Analysis-3S (RESAR-3S)", Volumes 1-8, July, 1975.
2. Amendments 1-10 to RESAR-3S.
3. USNRC, "Report to the Advisory Committee on Reactor Safeguards in the Matter of Westinghouse Electric Corporation Reference Safety Analysis Report, RESAR-3S," May 25, 1976.

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

June 11, 1976

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

Subject: REPORT ON SWESSAR-P1, STONE AND WEBSTER ENGINEERING CORPORATION BALANCE-OF-PLANT DESIGN AS APPLIED TO COMBUSTION ENGINEERING, INC. CESSAR-80

POOR ORIGINAL

Dear Mr. Rowden:

At its 194th meeting, on June 3-5, 1976, the Advisory Committee on Reactor Safeguards reviewed the application of the Stone and Webster Engineering Corporation for a Preliminary Design Approval of its SWESSAR-P1, a standardized nuclear balance-of-plant (BOP) design that would interface with a single unit Combustion Engineering, Inc. CESSAR-80 pressurized-water nuclear steam supply system (NSSS). A similar review for a Westinghouse RESAR-41 design was conducted at the 190th meeting of the Committee and was discussed in its report of February 11, 1976. The description of SWESSAR-P1 provided in the February 11, 1976 report is applicable to CESSAR-80; the latter was reviewed and a report provided by the Committee on September 17, 1975. During its review, the Committee had the benefit of discussions with representatives of the Stone and Webster Engineering Corporation and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

The arrangement of SWESSAR-P1 provides extensive physical separation of critical safety-related equipment to protect against common mode failures associated with fires or other operational contingencies. However, complete design details for SWESSAR-P1 have not been developed and the concept has not yet been applied to a complete nuclear power plant design. Consequently, further review of the physical separation arrangement should be made prior to the Final Design Approval or when SWESSAR-P1 is proposed for a nuclear power plant for which a construction permit is being sought. The Committee wishes to be kept informed.

A matter of major concern in the NRC Staff's review has been the safety-related interfaces between the SWESSAR-P1 BOP design and the CESSAR-80 NSSS design, on one hand, and the custom-designed site-related structures and components, on the other hand. The responsibilities and requirements related to the SWESSAR-P1/CESSAR-80 interfaces have been partially defined in the Safety Analysis Reports for these two standardized designs. The Committee believes that these interface requirements are satisfactory for a Preliminary Design Approval, but expects the NRC Staff and the Applicant to continue to examine them further in connection with the proposal to use these designs for an actual plant when it is reviewed for a construction permit. The interfaces between SWESSAR-P1 and the site-related features are defined in the SWESSAR-P1 Safety Analysis Report, but have not yet been subjected to the test of a complete design for a nuclear power plant. The NRC Staff should review these interfaces in greater depth when a construction permit application is received.

The Committee recommends that, during the design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be performed to assure complete functional compatibility across each interface for the entire spectrum of anticipated operations and postulated design basis accident conditions.

The coordination of interdependent instrumentation and controls in the nuclear island and in the balance of plant will require attention at the time when SWESSAR-P1 is used as a portion of a nuclear power plant license application. These matters should be included in the NRC Staff's standard review plans.

The proposed orientation of the turbine-generator with respect to the nuclear island is suitable for a single unit installation. For multiple unit power plants the location and orientation of the units should be such as to yield acceptably low probabilities of damage by low-trajectory turbine-generator missiles, or suitable missile shielding should be provided.

The SWESSAR-P1 and the CESSAR-80 NSSS designs, as do many others, utilize the concept of two-track continuous duty systems which perform critical service functions. In some cases the probability of failure of one of these systems is not low. The failure of the second system to start or run may cause progressively damaging consequences. The Committee recommends that failures of this kind be evaluated to determine if the necessary reliability exists for these systems and whether remedial measures are appropriate.

Although SWESSAR-P1 and CESSAR-80 include provisions for protection against industrial sabotage, the Committee believes that further steps can be taken beyond those provided. Prior to the use of SWESSAR-P1/CESSAR-80 as a portion of an application for a nuclear power plant license, the Utility-Applicant should be required to demonstrate that acceptable industrial sabotage provisions will be incorporated into the plant design.

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The SWESSAR-P1 design includes some provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. However, when SWESSAR-P1 is used as a portion of a nuclear power plant license application the Committee believes that the NRC Staff and the Applicant should further review methods and procedures for removing accumulated radioactive contamination whereby maintenance and inspection programs and ultimate decommissioning can be more effectively and safely carried out.

Generic problems related to large water reactors are discussed in the Committee's report dated April 16, 1976. Those problems relevant to SWESSAR-P1 and CESSAR-80 should be dealt with appropriately by the NRC Staff and the Applicant as solutions are found.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during the standardized plant licensing process and that, if due consideration is given to the foregoing and to the recommendations in the Committee's report of September 17, 1975 on CESSAR-80, Preliminary Design Approval for SWESSAR-P1 to be used in conjunction with CESSAR-80 can be granted in accord with the spirit and purposes set forth in the Commission's policy statement on standardization of nuclear power plants as described in WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants" and in conformance with the Regulations of Appendix O to Part 50 and Section 2.110 of Part 2 of Title 10 of the Code of Federal Regulations.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

POOR ORIGINAL

References

1. Pressurized Water Reactor Reference Nuclear Power Plant Safety Analysis Report (SWESSAR-P1) and Amendments 1 through 25.

2. Stone and Webster Engineering Corporation letters:
   a. January 12, 1976 - Responses to Outstanding Issues
   b. February 18, 1976 - Design Load Rejection Capability

3. Report to the Advisory Committee on Reactor Safeguards in the Matter of Stone and Webster Engineering Corporation Standard Safety Analysis Report PWR Reference Nuclear Power Plant SWESSAR-P1 (and its relationship to the CESSAR Standard NSSS Design) Docket No. STN 50-495, Published: May 1976, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 11, 1976

Honorable William A. Anders
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON SWESSAR-P1, STONE AND WEBSTER ENGINEERING CORPORATION BALANCE-OF-PLANT DESIGN

Dear Mr. Anders:

At its 190th meeting on February 5-7, 1976, the Advisory Committee on Reactor Safeguards reviewed the application of the Stone and Webster Engineering Corporation for a Preliminary Design Approval of its SWESSAR-P1, a standardized nuclear balance-of-plant (BOP) design. SWESSAR-P1 had previously been reviewed at Subcommittee meetings held in Chicago, Illinois, on August 1, 1975, and in Washington, DC, on January 22, 1976. During its review, the Committee had the benefit of discussions with representatives of the Stone and Webster Engineering Corporation and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

The SWESSAR-P1 BOP design would interface with single unit pressurized-water-reactor nuclear islands of standardized design such as RESAR-41 and CESSAR-80. This review is limited to RESAR-41. The interface requirements with other standardized nuclear island designs have not yet been established.

The SWESSAR-P1 containment is a conventional reinforced-concrete-steel-lined building with a flat base, a cylindrical shell, and a hemispherical dome. It is surrounded by an annulus building extending about one-half the height of the containment building. The containment and the annulus buildings are supported on a common base mat. The annulus building contains portions of the engineered safety features and some auxiliaries. The turbine generator is housed in a separate turbine building with its axis oriented radially with respect to the containment structure. Separate buildings are provided to house the diesel generators, the control facilities, and the radioactive waste disposal equipment.

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A set of design parameters has been established for SWESSAR-P1 which enable it to be applied to a range of sites without site-specific-design treatment of many features. The design was reviewed for a power level of 3800 MWt and would accept seismic loadings equivalent to 0.30g horizontal ground acceleration for the safe shutdown earthquake (SSE) and 0.15g horizontal ground acceleration for the operating basis earthquake (OBE). SWESSAR-P1 would be usable under meteorologic conditions prevailing in 22% of the more than 40 sites reviewed in this context by the NRC Staff. An optional extension of the annulus building to enclose the entire containment structure would permit SWESSAR-P1 to accommodate meteorological conditions at most sites thus far licensed. Other site conditions such as tornado design requirements, missile resistance, flood design limits, and postulated pipe rupture effects inside and outside containment are comparable to those now being required in licensed nuclear power plants. The remaining related design features such as offsite power, ultimate heat sink, and condenser cooling water supply and return, would be individually selected to suit the site on which SWESSAR-P1 is used.

The arrangement of SWESSAR-P1 provides extensive physical separation of critical safety-related equipment to protect against common mode failures associated with fires or other operational contingencies. However, complete design details for SWESSAR-P1 have not been developed and the concept has not yet been applied to a complete nuclear power plant design. Consequently, further review of the physical separation arrangement should be made prior to the Final Design Approval or when SWESSAR-P1 is proposed for a nuclear power plant for which a construction permit is being sought. The Committee wishes to be kept informed.

A matter of major concern in the NRC Staff's review has been the safety-related interfaces between the SWESSAR-P1 BOP design and the RESAR-41 NSSS design, on one hand, and the custom-designed site-related structures and components, on the other hand. The responsibilities and requirements related to the SWESSAR-RESAR interfaces have been defined in detail in the Safety Analysis Reports for these two standardized designs. The Committee believes that these interface requirements are satisfactory for a Preliminary Design Approval, but expects the NRC Staff and the Applicant to continue to examine them further in connection with the proposal to use these designs for an actual plant when it is reviewed for a construction permit. The interfaces between SWESSAR-P1 and the site-related features are defined in the SWESSAR-P1 Safety Analysis Report, but have not yet been subjected to the test of a complete design for a nuclear power plant. The NRC Staff should review these interfaces in greater depth when a construction permit application is received.

The Committee recommends that, during the design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be performed to assure complete functional compatibility across each interface for the entire spectrum of anticipated operations and postulated design basis accident conditions.

The coordination of interdependent instrumentation and controls in the nuclear island and in the balance of plant will require attention at the time when SWESSAR-P1 is used as a portion of a nuclear power plant license application. These matters should be included in the NRC Staff's standard review plans.

The proposed orientation of the turbine-generator with respect to the nuclear island is suitable for a single unit installation. For multiple unit power plants, the location and orientation of the units should be such as to yield acceptably low probabilities of damage by low-trajectory turbine-generator missiles, or suitable missile shielding should be provided.

Although SWESSAR-P1 includes provisions for protection against industrial sabotage, the Committee believes that further steps can be taken beyond those in SWESSAR-P1 and in the custom plant designs about which the ACRS has previously expressed concern. Prior to the use of SWESSAR-P1 as a portion of an application for a nuclear power plant license, the Utility-Applicant should be required to demonstrate that acceptable industrial sabotage provisions will be incorporated into the plant design.

The SWESSAR-P1 design should include provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. In particular, the Committee believes that the NRC Staff and the Applicant should review methods and procedures for removing accumulations of radioactive contamination whereby maintenance and inspection programs can be more effectively and safely carried out.

Generic problems related to large water reactors are discussed in the Committee's report dated March 12, 1975. Those problems relevant to SWESSAR-P1 should be dealt with appropriately by the NRC Staff and the Applicant as solutions are found.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during the standardized plant licensing process and that, if due consideration is given to the foregoing, Preliminary Design Approval for SWESSAR-P1 to be used in conjunction with RESAR-41 can be granted in accord with the spirit and purposes set forth in the Commission's policy statement on standardization of nuclear power plants as described in WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants" and in conformance with the Regulations of Appendix 0 to Part 50 and Section 2.110 of Part 2 of Title 10 of the Code of Federal Regulations.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

References

1. Pressurized Water Reactor Reference Nuclear Power Plant Safety Analysis Report (SWESSAR) and Amendments 1 through 20.

2. Stone and Webster Engineering Corporation letters:
   a. April 8, 1975 - Containment and Subcompartment Analysis
   b. April 18, 1975 - Subcompartment Analysis
   c. April 29, 1975 - Schedules
   d. April 30, 1975 - Steam Pipe Break Analysis
   e. June 4, 1975 - Implementation of WASH-1341
   f. June 5, 1975 - Supplementary Leak Collection and Release System
   g. September 5, 1975 - Reactor Cavity Nodulization Study
   h. September 5, 1975 - Schedules
   i. September 11, 1975 - Electrical System
   j. September 29, 1975 - Boron Recovery System
   k. October 2, 1975 - Interface Data
   l. November 13, 1975 - Supplementary Leak Collection and Release System
   m. November 21, 1975 - Resolution of Outstanding Items
   n. November 26, 1975 - Electrical, Instrumentation and Control Systems
   o. December 9, 1975 - Soil-Structure Interaction

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Reference - Continued

3. Report to the Advisory Committee on Reactor Safeguards in the Matter of Stone & Webster Engineering Corporation Standard Safety Analysis Report - PWR Reference Nuclear Power Plant SWESSAR-P1 (and its relationship to the RESAR-41 Standard NSSS Design) Docket No. STN 50-495, Published: October 1975, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

4. Supplement No. 1 to the Report to the Advisory Committee on Reactor Safeguards in the Matter of Stone & Webster Engineering Corporation Standard Safety Analysis Report PWR Reference Nuclear Power Plant SWESSAR-P1 (and its relationship to the RESAR-41 Standard NSSS Design) Docket No. STN 50-495, January 1976, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

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

September 17, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Anders:

Subject: COMBUSTION ENGINEERING STANDARD SAFETY ANALYSIS REPORT - CESSAR-80

At its 185th Meeting, September 11-13, 1975, the Advisory Committee on Reactor Safeguards completed its review of the Application of Combustion Engineering, Inc. for a Preliminary Design Approval (PDA) for its Standard Reference System-80, Safety Analysis Report CESSAR-80. Subcommittee meetings were held with representatives of the Applicant, and the Nuclear Regulatory Commission (NRC) Staff in Windsor, Connecticut, on February 28 and March 1, 1975, and in Washington, D. C., on May 23 and July 25, 1975. The full Committee met with representatives of the NRC Staff and the Applicant at its 184th Meeting August 14-16, 1975. The Committee also had the benefit of the documents listed below.

The Reference System-80 design consists of the nuclear steam supply system (NSSS) with a rated core power of 3800 MW(t), the NSSS control system, reactor protection system, engineered safety features actuation system, chemical and volume control system, shutdown cooling system, safety injection system and fuel handling system. Combustion Engineering will provide, at the option of the user, certain other safety-related systems which are outside the scope of the Reference System-80 design. These non-standard systems will be dealt with in the user's Safety Analysis Reports.

The Reference System-80 has been designed for application to an envelope of plant sites which encompasses all sites approved to date for Combustion Engineering NSSS. CESSAR-80 provides seismic response spectra for all major components, and equipment and piping systems, and other information required to ensure that the balance of plant is designed to protect the Reference System-80 from all site-related hazards. Application of the Reference System-80 design will require an evaluation of each site to confirm its acceptability within the CESSAR-80 envelope. For multiple reactor units at a single station, CESSAR-80 requires that each important safety-related item of the Reference System-80 design be provided for each reactor unit.

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CESSAR-80 will provide safety-related interface requirements information essential to the design of the balance of plant consistent with the assumptions used by Combustion Engineering in its accident analyses. Since the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related design requirements have been met, the Committee will review these matters in more detail with the utility-applicants on a case-by-case basis. The Committee recommends that, during the design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for an entire spectrum of anticipated operations and postulated design basis accident conditions.

The NRC Staff has identified several outstanding issues which will require resolution before the issuance of the PDA. The Committee recommends that these matters be resolved in a manner satisfactory to the Staff. The Committee wishes to be kept informed on the resolution of the following issues:

1. The emergency core cooling system evaluation.

2. The analysis of the effects of anticipated transients without scram.

3. Generic review of the effects of failures of reactor pump lubrication oil and component cooling water supply systems.

The most recent ACRS reports on nuclear power stations utilizing Combustion Engineering NSSS are the December 12, 1974 report on the application to construct the 2570 MW(t). St. Lucie Plant, Unit No. 2 and the June 10, 1975 report on the application to operate the 2570 MW(t). St. Lucie Plant, Unit No. 1. The Committee report on the 3390 MW(t). San Onofre Nuclear Power Generating Station, Units Nos. 2 and 3, selected by the Staff for reactor system design comparison with the Reference System-80 design, was issued July 21, 1972. Generic matters which include possible pump overspeed during a loss of coolant accident, transients associated with inadvertent operation of the emergency core cooling system or chemical and volume control system charging pumps, and analyses of postulated ruptures of the steam generator feed line, should be dealt with appropriately by the Staff. With regard to the rupture accident, the Committee recommends that the Staff perform an independent check on the calculation of steam generator blowdown force effects. It is expected that these items will be resolved in a manner satisfactory to the NRC Staff following the PDA and prior to the Final Design Approval (FDA). During the interim period, the Committee will continue to review these items on a case-by-case basis as well as through other appropriate Subcommittee and full Committee meetings.

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The peak linear heat generation rate is reduced to 12.1 kw/ft in order to meet the ECCS final acceptance criteria of Appendix K, 10 CFR 50. The Committee recognizes that conservative restrictions used in the NRC-approved ECCS model and the use of a generalized containment envelope yielding low containment pressures may be factors contributing to the imposed reduction in the permissible linear heat generation rates. The reduced limit imposes restrictions on modes for plant operation and becomes dependent on in-core monitoring systems for verification that limits are not exceeded. The Committee recommends that for a standard reactor of this size, larger safety margins, such as obtainable from higher reflooding rates, should be demonstrated. Programs underway by Combustion Engineering, Inc., include analytical and experimental studies aimed at providing the technical base for ECCS model improvements, as well as studying possible changes involving augmented ECC systems. The Committee believes that these programs constitute a sufficient basis for proceeding at this time and that the demonstration of larger safety margins should be part of the first major revised version of the Reference System-80 design which, as stated by Combustion Engineering, Inc., is likely to be submitted for review in about two years.

The Committee needs to complete its review of the suitability of the new 16 x 16 fuel and modified core reactivity controls of the Reference System-80 design which are now scheduled for initial proof testing at Arkansas Nuclear One, Unit No. 2 and at St. Lucie Plant, Unit No. 2. The Committee also needs to complete its review of the new core protection calculator system and the computer-based core operating limit supervisory system which will be incorporated into the Reference System-80 design in the event they are successfully demonstrated at Arkansas Nuclear One, Unit No. 2. The Committee needs to be assured of the dependability of in-core neutron flux sensors for control of reactors operating at low core power peaking factors. For this purpose the Committee recommends that the Staff and the Applicant continue to gather pertinent information from operating CE reactors. The Committee will continue its review of these matters as appropriate documentation is submitted and the improvements sought can be evaluated.

The Committee recognizes the importance to safety and improved designs of developing computational methods to provide best estimate analyses of LOCA and other postulated accidents. The Committee encourages the Applicant and the NRC Staff to accelerate their efforts to this end. The Committee wishes to be kept informed.

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The CESSAR-80 design should include provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. In particular, the Committee believes that the NRC Staff and Combustion Engineering, Inc., should review methods and procedures for removing accumulations of radioactive contamination whereby maintenance and inspection programs can be more effectively and safely carried out.

The Committee believes that Combustion Engineering and the NRC Staff should continue to review the Reference System-80 for design changes that will further improve protection against sabotage.

The Committee believes that methods that seek to develop reference systems through standardization and through replication need to be coupled with ongoing programs that will permit design changes to reference systems which improve safety and which, when justified, will be implemented in a timely manner. Use of reference systems should lead to more efficient and effective licensing reviews. Programs such as CESSAR-80 will contribute to this process. A transition period will be required in which the Committee will still give attention to the items noted, on a case-by-case basis.

The Committee believes that, subject to the above comments and successful completion of the R&D programs, the Combustion Engineering Reference System-80 design can be successfully engineered to serve as a reference system.

Sincerely yours,

WKerr

William Kerr
Chairman

POOR ORIGINAL

REFERENCES TO THE CESSAR-80 LETTER:

1. Combustion Engineering Standard Safety Analysis Report for System-80 (CESSAR with Amendments 1 through 36

2. Report to the Advisory Committee on Reactor Safeguards from the Office of Nuclear Reactor Regulation, dated July, 1975

3. Supplement 1 to the Report to the Advisory Committee on Reactor Safeguards from the Office of Nuclear Reactor Regulation, dated August 8, 1975

4. Letter, dated March 24, 1975, Combustion Engineering, Inc., to DRL concerning information on the fuel transfer tube

5. Letter, dated March 10, 1975, Combustion Engineering, Inc., to DRL concerning radioiodine spiking effects on accident releases

6. Letter, dated January 15, 1975, Combustion Engineering Inc., to DRL concerning views on Anticipated Transients Without Scram

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

March 14, 1975

Honorable William A. Anders
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT (GESSAR-238)

Dear Mr. Anders:

At its 179th Meeting, March 6-8, 1975, the Advisory Committee on Reactor Safeguards completed a review of the General Electric Standard Safety Analysis Report (GESSAR). GESSAR-238 provides the safety information for a reference system consisting of a single BWR-6/Mark III nuclear system, with a rated core thermal power of 3579 MW(t), and of the associated systems including the reactor building (the shield building and containment), fuel building, auxiliary building, diesel generator buildings, control building, radwaste building,, and the off-gas system. Subcommittee meetings were held with representatives of the General Electric Company and the Nuclear Regulatory Commission (NRC) Staff on July 1, 1974, and September 11, 1974, in Washington, D. C., on November 9, 1974, in Bloomington, Minnesota, and on January 13, 1975, in Washington, D.C. The Committee also had the benefit of the documents listed below.

Site envelope parameters are included in GESSAR and application of GESSAR will require that specific site evaluations be made to confirm the acceptability of the site within the GESSAR design. The use of GESSAR for multiple reactor units at a single station will also require review of the safety-related components of plant duplication and layout.

Safety-related interfaces between the reference system and the balance of plant are specified in GESSAR. Since the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related interfaces have been identified and that all safety-related requirements are being fulfilled, the Committee will review these matters in more detail with the Applicants on a case-by-case basis. The Committee recommends that, during the design, procurement, construction and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for an entire spectrum of anticipated operations and postulated design basis accident conditions.

The NRC Staff has identified 13 items requiring resolution prior to issuing their Preliminary Design Approval (PDA). The Committee believes that all of these matters should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed regarding the resolution of the following items:

1. Seismic capability of the offgas system.

2. Provisions to satisfy the single-failure criterion for the RHR system.

3. Additional requirements to be imposed if continuous venting of the containment is used.

4. Evaluation of the performance of the emergency core cooling systems using evaluation models meeting the requirements of 10 CFR 50.46, Appendix K.

The latest ACRS reports on nuclear generating stations utilizing the BWR-6/Mark III systems were the December 12, 1974 reports on the Allens Creek Nuclear Generating Station, Units 1 and 2, and the Perry Nuclear Power Plant, Units 1 and 2. In these reports, the ACRS has recommended that the ongoing R&D programs be used to fully resolve issues involving the Mark III containment design prior to completion of the affected portions of the plant. Further, additional generic matters, which include anticipated transients without scram (ATWS) and possible pump overspeed during a loss of coolant accident, should be dealt with appropriately by the NRC Staff. It is expected, that these items will be resolved in a manner satisfactory to the NRC Staff following Preliminary Design Approval (PDA) of GESSAR and prior to Final Design Approval (FDA). During this interim period, the Committee will continue to review these items on a case-by-case basis as well as through other appropriate ACRS Subcommittee meetings and full Committee meetings.

The Committee has not reviewed modifications which are expected to be made in the BWR/6 8x8 fuel. Such modifications and any other proposed changes will be reviewed when the appropriate documentation has been submitted and the improvements sought can be evaluated.

The introduction of new features in the instrumentation and control systems has been submitted through the specification of functional designs and design criteria which the NRC Staff has found to be adequate for the PDA. As in previous reports on related matters the Committee recommends that the NRC Staff determine the necessary environmental and reliability tests, including in situ tests where desirable for qualification of the new systems. In another matter relating to a periodic testing provision, the General Electric Company has committed to a study of the improvement of the testability of the automatic depressurization system. On all these issues involving instrumentation and control, the Committee will use the case-by-case basis to ascertain progress of the work until the GESSAR design has progressed to the stage where Final Design Approval is achieved.

The Committee will need to review the development and proof testing of the fast scram system, and the implementation of the proposed Reactor Manual Control System along with the provisions for ganged rod withdrawal.

The Committee believes that the General Electric Company and the NRC Staff should continue to review GESSAR for design changes that would further improve industrial security features.

The GESSAR design should include provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. In particular, the Committee believes that the NRC Staff and the General Electric Company should review methods and procedures for removing accumulations of radioactive contamination whereby maintenance and inspection programs can be more effectively and safely carried out.

The Committee believes that methods that seek to develop reference systems through standardization and through replication need to be coupled with ongoing programs that will permit changes which improve safety and which, when justified, would be implemented in a timely manner. Use of reference systems should lead to more efficient and effective licensing reviews. Programs such as GESSAR will contribute to this process. A transition period will be required in which the Committee would still give considerable attention to the items noted, on a case-by-case basis.

The Committee believes that, subject to the above comments and to successful completion of the R&D prgrams, GESSAR-238 can be successfully engineered to serve as a reference system.

Sincerely yours,

WKerr

William Kerr
Chairman

POOR ORIGINAL

References attached.

References

1. BWR/6 Standard Safety Analysis Report, Volume 1 through 7.

2. Amendments 1 through 28 to the Standard Safety Analysis Report.

3. General Electric Company letters and reports:

   a. July 31, 1973 letter forwarding proprietary information in support of the information made public in the safety analysis report.

   b. August 31, 1973 letter forwarding proprietary fuel data.

   c. September 28, 1973 letter forwarding proprietary information regarding core power distribution.

   d. December 28, 1973 letter regarding interfaces and electrical systems.

   e. November 6, 1974 letter regarding physics verification and number of safety/relief valves.

   f. February 19, 1974 letter regarding ATWS.

4. AEC/NRC Staff letters and reports:

   a. October 11, 1974 draft Safety Evaluation Report.

   b. November 13, 1974 Safety Evaluation Report.

   c. December 7, 1974 Supplement No. 1 to the Safety Evaluaion Report.

   d. January 30, 1975 letter regarding reevaluation of the high pressure drywell test.

   e. February 21, 1975 Supplement No. 2 to the Safety Evaluation Report.

   f. March 4, 1975 Supplement No. 3 to the Safety Evaluation Report.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 8, 1974

L. M. Muntzing
Director of Regulation

SYSTEMS ANALYSIS OF ENGINEERED SAFETY SYSTEMS

With the current effort to standardize the design of certain types of nuclear power plants, the Committee believes that attention to the evaluation of safety systems and associated equipment from a multidisciplinary point of view to identify potentially undesirable interactions between systems becomes increasingly important. The attached illustrative examples represents an initial and not necessarily complete listing of some problem areas.

The Committee would appreciate the Regulatory Staff reviewing these comments and discussing their ideas with an appropriate Subcommittee. Based on these discussions a mutually beneficial procedure for handling such issues may be developed.

W. R. Stratton
Chairman ACRS

Attachment:
List of Illustrative Examples

cc: P. Bender, SEC
E. Case, DL

COPY

Illustrative Examples of Questions to be answered by Systems Analysis and Quality Assurance

The following comments and questions are suggested for consideration as additional guidance in the review of Engineered Safety Systems:

1. Comment: Designers and architect-engineers frequently delegate responsibility for systems analyses to teams with functional engineering specialties such as "civil," "electrical," "mechanical," or "nuclear" with the team effort coordinated by managers responsible for controlling costs and avoiding schedule delays. With the same standard design applied to a number of plants, an intensive systems analysis effort which integrates the functional engineering specialties, is feasible. The scope and approach of the related Quality Control, Quality Assurance effort should be commensurate with the Project Design effort. Consideration should be given to identifiable multi-disciplinary analyses of safety-related systems and associated systems, as part of Quality Control in design, procurement, construction, operation, and maintenance activities.

   General Question:

   What are the respective roles played by Project Design and Quality Assurance/Quality Control in the multi-discipline analyses of safety systems and associated systems?

2. Comment: As an aid in identification, safety related systems and associated equipment may be categorized as follows:

   a) Those systems or items of equipment which must be de-energized on demand (to a zero energy state) with extremely high reliability to:

      (1) Perform a safety function;

      (2) Prevent fire or other damaging consequence.

   b) Those systems which must be capable of long-term active operation to preserve control over radioactive materials (examples are fuel and environmental cooling and lighting and communications services).

   c) Those systems not directly related to a safety function but whose malfunction could have safety consequences because of secondary effects. It should be noted that such systems may not ordinarily be included in the set for which "conditions of design" are defined.

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Question: In the design of such systems, is an interdisciplinary systems analysis performed to assure redundancy and separation appropriate to the category of the system? Does it consider all modes of normal operation, operation following any of the design basis events plus additional incidents such as pipe failures, loss of all active inputs to the system, and operation of part of the active components combined with the failure of others (for example, the operation of a large and critical motor in a space where the ventilation has failed)?

3. Comment: In addition to systems and equipment, space allocation and arrangement are crucial to safety. Both Unit and Station systems must be analyzed to assure adequate independence and separation of all vital functions. The analysis should consider the possibility that adverse "feedback" or other effect from one unit may leave other units without adequate redundancy.

Such an analysis should help to provide a basis for establishing reliability, redundancy, and separation requirements. It should also provide information concerning the degree of separation necessary to protect against mechanical damage, fire or sabotage.

Questions:

a) Are design efforts and systems analyses directed to avoid concentration of vulnerability from various causes in one safety class structure, room, or zone?

b) Are field located and field run equipment and systems examined to see if localized vulnerability has been created?

c) Is space allocation a conscious responsibility in design?

d) Are field inspections of space occupied by safety equipment and systems made by the cognizant design engineer to assure non-encroachment?

e) Do changes in space allocation or arrangement require special approval?

4. Comments: Control systems may require communication lines, (electrical, pneumatic or hydraulic) that traverse significant distances and pass through several compartments.

Questions:

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a) Is attention given in system design to physical locations of "field-run" impulse or static lines, including lines that provide information regarding ECCS functions, and other electrical-mechanical-hydraulic-pneumatic control systems which perform safety functions, to assure that an unacceptable interaction between these and other systems is avoided?

b) Is specific attention given to assuring that field location follows that specified in the design?

5) Comments: Electrical systems and equipment should be analyzed to assure that over-current or other fault protection is sufficiently reliable and redundant to assure appropriate limitation of damage potential to other safety systems.

An example is the electrical power supply to primary system pumps. Failure of the circuit breakers could result in damage to the electrical penetrations and loss of containment under post-LOCA conditions.

Questions:

a) Are the circuit breakers for electrical power circuits that pass through containment penetrations set to trip in the event of arcing faults within the penetration?

b) Are such circuits designed with ground fault trips to protect the penetrations?

c) Are ground fault trips provided on all power circuits within the physical safety complex to reduce the fire hazard?

d) Are emergency lighting systems and internal communication systems safety grade?

e) Are control and power cables of widely differing voltages and currents intermixed in cabletrays, raceways, or conduits?

f) Are magnetic forces and molten copper considered in specifying the separation required between cables?

g) Are differences between laboratory test conditions for flame resistant cable insulation, and conditions that could exist in a cable way under faulted conditions, considered in defining separation requirements?

h) In determining the adequacy of separation, is consideration given to "foreign" sources of damage such as vehicle impact, use of welding equipment, explosive gas accumulation, or acts of sabotage?

i) Is fireproof, rather than fire "retardant" insulation required in vital areas? Is potential damage from radiation exposure from nearby components, such as air filters and charcoal adsorption beds, taken into account?

j) Is the timing of loss of offsite power considered in the prediction of the consequences of an accident? (For example, the most disadvantageous time may be just as motor operated valves are about to open or large pumps are almost up to operating speed).

6. Comments: Some ventilation systems may not be given attention as engineered safety features, however, situations may arise in which they can have important effects on safety.

Questions:

a) Are auxiliary systems such as containment or reactor building air cooling systems analyzed to see if their failure can lead to the failure of safety systems?

b) Are dynamic as well as static differential pressures on containment ventilation ducts and isolation dampers considered?

c) Are local effects of flow pressure gradients resulting from pipe ruptures analyzed for phenomena such as the collapsing of ventilation ducts, which could result in closing vent areas?

d) In evaluating the adequacy of the protection provided to operators in the control room following a LOCA, is consideration given to the possibility that a ventilation (or large electrical) penetration of the containment has failed and is leaking the containment atmosphere into the adjacent space?

7. Comments: Experience has indicated that fluid systems deserve special attention in both static and dynamic situations. Particular attention should be given to stresses resulting from valve action, pump starts, and water slugs, including backflow and check valve action, as well as flow action under severe accident conditions or fault modes.

Questions:

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a) Is consideration given to the effect of fluid system dynamics on mechanical stresses in components and equipment?

b) Are the consequences of the failure of check valves to close properly in various fluid systems examined for normal and faulted conditions?

c) Is evaluation made of the possibility and effects of crushing and/or rupturing one group of control rod drive hydraulic lines during a LOCA? Are combinations of ruptured and crushed lines also considered?

d) In PWRs, are the consequences of multiple steam generator blowdown considered?

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e) In the evaluation of a system's ability to perform its required service is consideration given to potential flooding effects resulting from roof drain obstruction (potential roof collapse), rupture of non-Class I tankage, continued operation of a leaking system, or reverse flow through normal or ruptured pipe that could be siphoning liquid from some storage source?

f) What controls are placed on the use of "plaster" or glass wool type thermal insulation within containment, that could foul or possibly cause failure of ECC systems?

g) Is an analysis performed to determine when pool boiling would occur, if during refueling (with the reactor vessel head removed) both of the two canal cooling systems become disabled?

What would be its consequences? How long would the operators have to restore cooling?

8. Comments: Fires may have unusual consequences in reactor systems and deserve special attention.

Questions:

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a) How are fires analyzed for potential effects on safety?

b) Are the storage of flammable materials in vital spaces and the passage of flammable gases or liquids through vital spaces prohibited?

c) Are safety enclosures, including doors, for diesel generators designed to withstand a diesel runaway, fire, or combined fire-explosion?

d) Does analysis of electrically generated fires consider the following for each power circuit:

1) The change of a circuit short or overload in a circuit within a safety class structure?

2) The chance of a branch overload or short circuit followed by failure to clear the fault?

3) The chance of fire from circuit overheating at or below normal current load?

4) The possibility that fire will propagate to:

a) Disable one vital electrical division if the circuit is not already in a vital division?

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b) Disable two or more vital divisions in a local area where minimum allowable separation is employed?

5) The potential consequences of combustion of fumes from fire in confined spaces?

9. Comments: After careful analysis and design it is essential that operation or tests in the field follow the resulting specifications.

Questions:

a) Does environmental qualification allow for or test for the possible lack of discipline in field installation which may result in a field installation that is significantly different from the qualification test setup? Do the qualification tests represent a condition of long term (multi-year) normal operation followed by short term, very severe environmental conditions?

b) Are special instructions for operation and maintenance identified after being developed by a disciplined systems analysis? Examples of such instructions are "use no flame", "no traffic area", "do not operate if ...," "no welding without prior approval of fire protection personnel", "do not use mercury-containing instruments", "do not overtorque", "no substitutes for this material".

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6 copies

PART 6

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SYS. INTER

EXhibiT 1069

MAY 1978

Present Status ?

Generic Task No. A-17

MEMORANDUM FOR: Edson G. Case, Director, Office of Nuclear Reactor Regulation

FROM: Roger S. Boyd, Director, Division of Project Management

SUBJECT: SUBMITTAL OF REVISED TASK ACTION PLAN

Proposed Revision 1 to Task Action Plan A-17, "Systems Interaction in Nuclear Power Plants," is enclosed for review and approval by the Technical Activities Steering Committee (TASC). The plan has been extensively revised to reflect the technical assistance contract at Sandia Laboratories under joint management by NRR and OSD. This was discussed at the TASC Meeting No. 8 on February 3, 1978.

A draft of this proposed revision was circulated for comments on February 16, 1978. Comments from other Division Directors have been resolved or incorporated into the revised plan.

Original signed by:
Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosure:
Proposed Revision 1 to
Task Action Plan A-17

Distribution
Central Files
J. Angelo
M. Aycock
H. Balukjian
W. C. Burke
G. Chipman
M. Chirmal
L. Crocker
R. C. DeYoung
H. Denton
J. Guibert
H. Li
C. Liang
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H. Schierling
V. Stello
D. Wigginton
J. Zwolinski

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2-2 3-38 3-2 5-2 5-1 3-48 5-1 -29 5-2 4-2

TASK ACTION PLAN

TASK NO. A-17

TITLE: Systems Interaction in Nuclear Power Plants

LEAD RESPONSIBILITY: Division of Project Management

LEAD ASSISTANT DIRECTOR: R. C. DeYoung, Deputy Director, DPM

TASK MANAGER: John Angelo

1. Problem Description

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines such as civil, electrical, mechanical, structural, chemical, hydraulic, and nuclear, and into scientific disciplines such as geology, seismology, and meteorology. The reviews performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements will be met. These reviews include failure mode analyses to assure that the single failure criterion is met.

The NRC review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan which assigns primary and secondary review responsibilities to organizational units arranged by plant systems such as containment systems, reactor systems, etc., or by disciplines such as mechanical engineering, materials engineering, and structural engineering. Each element of the Standard Review Plan is assigned to an organizational unit for primary responsibility and, where appropriate, to other units for secondary responsibilities.

Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of our regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

The problem to be resolved by this task is to identify where the present design, analysis, and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures.

2. Plan for Problem Resolution

The plan for resolution of this task is to develop a method for conducting a disciplined and systematic review of nuclear power plant systems, for both process function couplings of systems and space couplings, to identify the potential sources and types of systems interactions that are determined to be potentially adverse. A set of criteria that will be developed early in the execution of this task to bound its scope. It is anticipated that a matrix of systems and interactions will be synthesized generically for a nuclear power plant and verified for a selected facility. This matrix could be displayed as plant logic and system models, for example, somewhat analogous to techniques that have already been developed for similar kinds of studies and analyses. The Standard Review Plan will then be measured against this synthesized matrix to determine the completeness of the review procedure. From this comparison, any necessary revisions to the review procedures, including necessary revisions to Standard Review Plans, Regulatory Guides, etc., will be developed and recommended for implementation.

The plan will also include the development of criteria and procedures to assure that applicants incorporate appropriate systems interaction considerations into their design and review process.

The plan is to be accomplished in two phases. Phase I will include the development of a systematic review process for plant systems and interactions and the verification of the Standard Review Plan against the results of the systematic review. This phase, is expected to be completed in 12 months following the assignment of manpower and funding resources. Throughout this phase any results that indicate a need for immediate regulatory action will be identified and appropriate recommendations made to management. A final report summarizing the results of Phase I will be issued at the completion of the phase. Phase II will be accomplished within approximately 12 months after the completion of Phase I. This phase will include the preparation of follow-on actions that are necessary to implement the results of this task. All of these follow-on actions will have been identified during Phase I. The follow-on actions include any necessary revisions to the Standard Review Plan, Regulatory Guides, or other regulatory actions. Since it is not possible at this time to specify what the nature, extent or scope of these follow-on actions might be, the detailed scheduling of Phase II cannot be completed until most of Phase I work has been accomplished.

The plan will be accomplished by the coordinated efforts of three groups: (1) a Systems Interaction Working Group composed of individuals selected from organizational branch units within the Office of Nuclear Reactor Regulation (NRR) that are impacted the most by plant systems interactions and accident analyses working with a Task Manager, (2) a group within the Office of Standards Development (OSD) working with a Project Manager in the Engineering Methodology Standards Branch of OSD, and (3) a group working for Sandia Laboratories under contract to OSD. Further details of this arrangement and the work to be accomplished by each group are provided in Sections 3, 4 and 6 of this Task Action Plan. Attachment 1 shows the lines of administrative and technical control that are proposed for the execution of this task.

The major elements of this Task are described in the following paragraphs:

(a) Sandia Laboratories will, through the accomplishment of the work described in Section 4 of this Task Action Plan, develop a systematic review process for systems interactions. Sandia Laboratories will verify and demonstrate the review process for an exemplary facility and will assess the Standard Review Plan against the systematic review process. All of this will be accomplished during Phase I of the Task. In Phase II Sandia Laboratores will, as determined to be appropriate and necessary, transfer the techniques of the review methodology to NRC for further use by the NRC staff and will provide assistance in the follow-on actions indicated by the results of Phase I of the Task. Sandia Laboratories will report the progress of work at periodic intervals as indicated in Section 7 of this Plan.

(b) The NRR Systems Interaction Working Group will review and evaluate the work performed by Sandia Laboratories and will provide assistance in specialized technical areas to supplement the technical capabilities of the group at Sandia Laboratories. The NRR Systems Interaction Working Group will also provide the evaluation needed to form the technical basis for NRR management decisions regarding the acceptability of the task efforts by Sandia Laboratories. The NRR group, through the Task Manager, will transmit its findings to the Office of Standards Development (OSD).

(c) The Office of Standards Development will administer and manage the contract with Sandia Laboratories through its assigned Project Manager. OSD will also provide technical review, evaluation and direction of the work performed by Sandia Laboratories in conjunction with the technical overview by the NRR System Interaction Working Group. It is recognized that this joint effort by NRR and OSD will require careful coordination between the Project Manager at OSD and the Task Manager at NRR.

(d) NRR cognizant branches will perform a review of each task accomplished by Sandia Laboratories. The review comments and evlauation will be forwarded by each of the cognizant branches to the NRR Task Manager. All such comments may be further considered by the NRR Systems Interaction Working Group to resolve any conflicts. The NRR cognizant branches include all of the branches within the Division of Operating Reactors, Division of Project Management and the Division of Systems Safety. The NRR cognizant branches also include the following branches in the Division of Site Safety and Environmental Analysis: Accident Analysis Branch, Effluent Treatment Systems Branch, Radiological Assessment Branch, Geosciences Branch, and Hydrology-Meteorology Branch.

Eight of the most heavily impacted NRR cognizant branches have an assigned member on the NRR Systems Interaction Working Group. This Task Action Plan allows these branches to make input to the task through the assigned branch representative or separately from the NRR Systems Interaction Working Group at the option of the individual branch chief.

To accomplish this Task and to establish a uniform basis for review by cognizant review branches, it will be necessary to develop criteria for bounding the extent of systems interaction. The criteria must define the items that will be retained in the matrix of systems and interactions; otherwise, the matrix will become unmanageable and the review will not proceed on a uniform basis. The criteria will serve as the basis to eliminate systems interactions of little or no safety significance. These criteria will be developed early in the execution of the task in order to give purposeful direction to the task and to its review.

One of the end products of this task will be additions, where necessary, to the Standard Review Plan to assure that our review procedures adequately address considerations for systems interaction. Another end product will be a recommendation that a Regulatory Guide or other appropriate documentation be issued to provide guidance on the criteria, procedures, and information required related to applicants' analyses and review of systems interaction.

During the accomplishment of this task, consideration will be given to the use of the end products for operating reactors. The method of accomplishing this objective will be by review of the task by the individual assigned from the Division of Operating Reactors (DOR) to the NRR Systems Interaction Working Group. Since some of the elements of this systems interaction task are common to the elements that have been and will be used in the Systematic Evaluation Program (SEP) currently being conducted by DOR, the assignment has been made from the SEP Branch. This individual will make his recommendations to Division of Operating Reactors management.

3. <u>NRR Technical Organizations Involved</u>

The conduct of the task shall be the responsibility of NRR. The strong OSD contribution will be recognized by having an OSD representative assigned as Project Manager working with the assigned NRR Task Manager. Technical interface between OSD and NRR shall be conducted by the Task Manager for NRR. The Task Manager shall also retain the responsibility through the Technical Activities Steering Committee for meeting objectives and schedules established for the task. The Task Manager, through the lead supervisor, shall also be responsible for defining and revising, as necessary, the objectives and schedules as would be done for any other Category A Task.

The technical branches of NRR that are most affected by systems interactions have each appointed a principal person to act as a point of contact between the Task Manager and the branch and to be the primary technical representative of the branch. For all other branches within NRR, the Task Manager will act through the Branch Chief. The branches most affected by system interactions are:

Auxiliary Systems Branch
Instrumentation & Control Systems Branch
Power Systems Branch
Containment Systems Branch
Effluent Treatment Systems Branch
Accident Analysis Branch
Reactor Systems Branch
Systematic Evaluation Branch

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The representatives of these branches shall form the NRR Systems Interaction Working Group working with the Task Manager and, as the occasion demands, working directly with members of OSD or Sandia

Laboratory. This group will provide consultation or direct technical contribution on selected problems in their areas of expertise. An adequate amount of time should be allocated by the branches to enable these people to perform this activity. For present planning purposes, it is estimate about 20% time input of one man per branch for the more heavily impacted branches to about 10% time for lesser impacted branches.

Based on an estimated time of twelve calendar months to accomplish Phase I, and an estimated time of twelve months to accomplish Phase II, the following allocations of manpower requirements for the principal branches which have assigned personnel for the NRR Systems Interaction Working Group should be made:

	Phase I	Man-Months Phase II	Total
Auxiliary Systems Branch	2.4	1.8	4.2
Instrumentation & Control Systems Branch	2.4	1.8	4.2
Power Systems Branch	2.4	1.8	4.2
Containment Systems Branch	1.2	0.9	2.1
Effluent Treatment Systems Branch	1.2	0.9	2.1
Accident Analysis Branch	1.2	0.9	2.1
Reactor Systems Branch	2.4	1.8	4.2
Systematic Evaluation Branch	1.2	0.9	2.1
Total	14.4	10.8	25.2

In addition to these individuals, virtually all technical branches within DSS, DSE, DOR, and DPM will be requested to review and critique the end products of Phase I and Phase II efforts and provide a nominal level of time for consultation in selected areas. The requirements of specific branches will vary as a function of their involvement with systems. This time is anticipated to require about 15 man-months and will vary from one-half man-week to four man-weeks per branch. This time will be expended over the span of the task at the specific milestones indicated in paragraph 7 of this report. An overall estimate of this review effort is shown in Attachment 2 to this plan.

In addition to the review and critique by cognizant review branches within NRR, the assistance of the AD for Reactor Safeguards, DOR, will be requested for consultant assistance to aid in using the techniques for plant and systems reviews that was developed by the workshop group for Industrial Security.

4. Technical Assistance Requirements

This task will be accomplished by assistance from Sandia Laboratories working under a contract that will be administered by the Office of Standards Development. The contract will cover a two phase effort expended over an estimated time of 24 months. The first phase will include Task 1 through Task 13 as described below. The first phase is estimated to be completed in 12 months at a cost of $440,000. The second phase is estimated to be accomplished in 12 months at a potential cost of about $200,000. This estimate for Phase II represents an "upper-bound" estimation. Actual requirements will be dependent on the results of Phase I.

The specific tasks to be accomplished by Sandia Laborabories are described in Attachment 3. Since one of the major tasks is to define the scope of the program in more definitive details, the tasks described in Attachment 3 should be treated as reflecting the initial thoughts of Sandia Laboratories.

At appropriate points during the execution of this task, and as the results become available, the results of the ongoing technical assistance program with Oak Ridge National Laboratory (ORNL) now being conducted by DOR will be used in the task. In order to accomplish this objective, cognizance of the ongoing technical assistance program will be developed and maintained by review of published information, attendance at meetings, and conferences with personnel who are active in the program in DOR and ORNL.

The scope of the task at ORNL is (1) to identify and evaluate the safety significance of possible interactions between control and protection systems, (2) provide recommendations for possible design modifications or operational requirements, (3) perform a detailed analysis, including a failure mode analysis, of auxiliary control systems specified by the NRC for the purpose of identifying any dependence between these systems and the reactor protection system, (4) assess the possibility of control system failures resulting in a challenge to the reactor protection system, and (5) evaluate the significance of adverse interactions between protection and control systems, and the capability of the reactor protection systems to mitigate the consequences resulting from these interactions or from control systems failures.

Manpower and funding estimates for this task at ORNL are 15 man-months of support effort during FY 1979 at a cost of $50,000.

5. Interactions with Outside Organizations

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

Meetings are anticipated with NSSS vendors, A/Es, and utilities to assess the extent to which these organizations conduct reviews and analyses for systems interaction, and to keep these organizations informed of our developments. It is not intended, however, to conduct a formal review process through these organizations. The intent is to develop a free exchange of information so that the task can take advantage of existing methods of review.

Commonwealth Edison Company has performed and will implement a somewhat limited systems interaction study for the Zion Station. The Commonwealth Edison Company study will consist of a detailed review of Licensee Event Reports of those events which have occurred that involve undesirable systems interactions. Both physical and electrical interactions will be covered in the event review but will be approached on a case-by-case basis rather than from a more general standpoint. We agree that this study should proceed, recognizing that it may or may not be the final effort for the Zion facility since additional techniques may be developed at a later time.

6. Assistance Requirements from Other NRC Offices

The Office of Standards Development shall manage the contract effort and shall also provide technical input to the task effort to (a) supplement the contract effort, (b) direct and evaluate the contract effort, and (c) interface with the technical and management efforts by

NRR. It is estimated that this effort by OSD will total 14.4 man-months during Phase I of the task. The Phase II effort has not been estimated and is dependent on the results of Phase I.

Assistance will be requested from the Probabilistic Analysis Branch, Office of Nuclear Regulatory Research, to provide consultant assistance in the detailed development and execution of this task action plan. It is estimated that this total assistance from RES will be about one man-month of effort. It is anticipated that this group can provide valuable insights into the task because of its involvement with the Reactor Safety Study (WASH-1400). Additionally, this group would be requested to review and critique the results of this task action plan.

7. Schedule for Problem Resolution

The following schedule is proposed for execution of this task:

(a) Assignment of NRR Personnel to the Systems Interaction Working Group — MARCH 1, 1978

(b) Definition of Specific Tasks for Contract Assistance — APRIL 1, 1978

(c) Contract Awarded — MAY 1, 1978

(d) First Contractor Report Submitted	AUGUST 1, 1978
(c) Review Comments to Task Manager from All Cognizant Branches	AUGUST 30, 1978
(f) Second Contractor Report Submitted	DECEMBER 1, 1978
(g) Review Comments to Task Manager from all Cognizant Branches	DECEMBER 30, 1978
(h) Third Contractor Report Submitted	FEBRUARY 1, 1979
(i) Review Comments to Task Manager from all Cognizant Branches	FEBRUARY 28, 1979
(j) Fourth Contractor Report Submitted	MAY 1, 1979
(k) Review Comment to Task Manager from all Cognizant Branches	MAY 30, 1979
(l) Phase I Final Report Issued	AUGUST 1, 1979
(m) Phase II Task Defined	JUNE 1, 1979

(n) Phase II Contract Awarded JUNE 1, 1979

(o) Phase II Completed MAY 1, 1980

8. Potential Problems

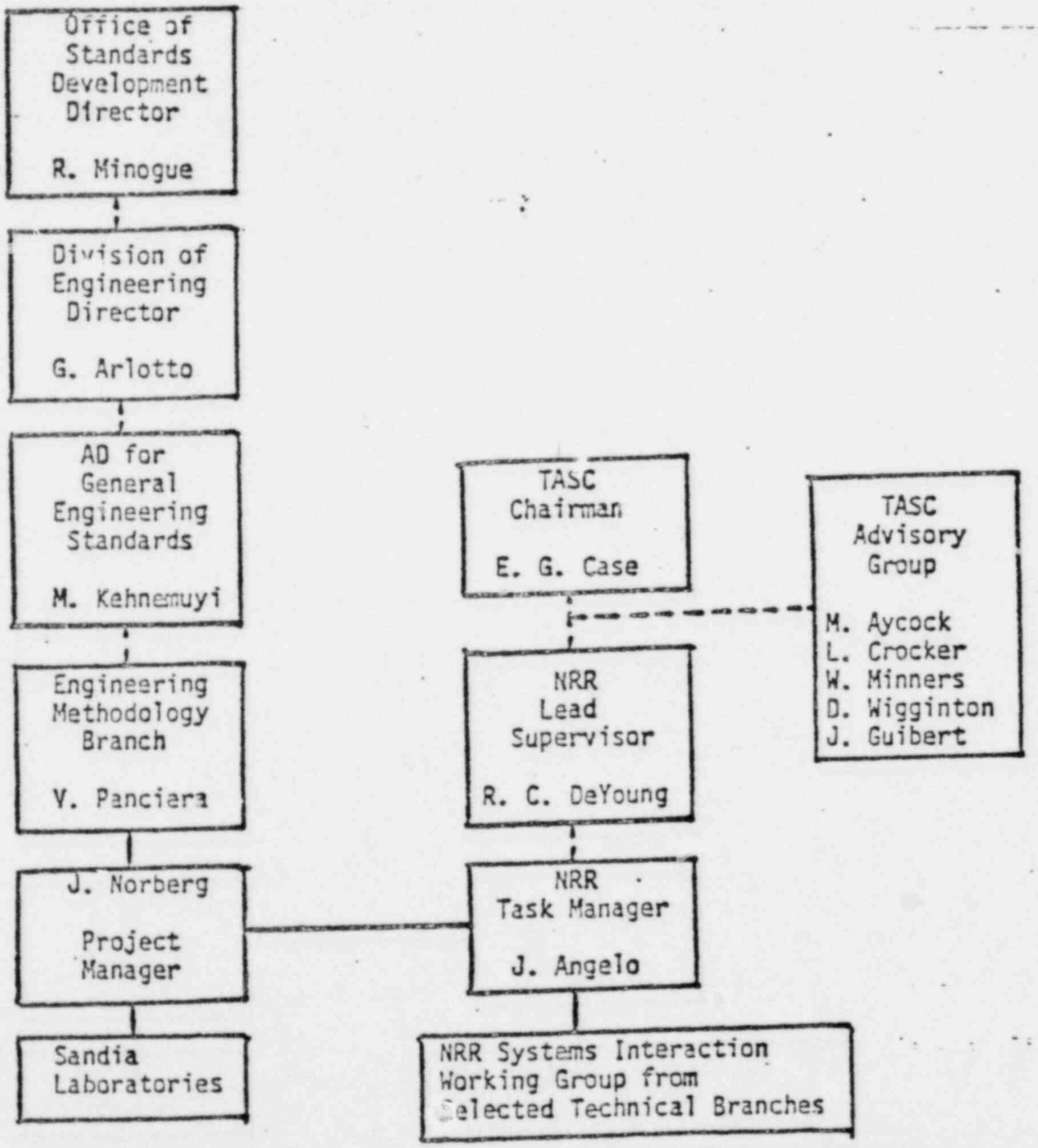
One of the problem areas is that systems interaction cuts across all disciplines and technical branch review areas and cuts across all groups and divisions. Consequently, in order to effectively perform a review for systems interaction, it is necessary to either define more clearly and more extensively the primary and secondary review responsibilities in the Standard Review Plan or organize a new element to perform the review. Consideration will be given during execution of this task to the resolution of this problem.

A second potential problem area is related to estimating the scope and extent of effort required to complete Phase II concerning the potential revisions to the Standard Review Plan and the development of criteria and procedures for use by applicants in their design and review of plant designs for systems interaction. Therefore, it is anticipated that at the completion of Phase I, a reassessment will be made of the follow-on effort. It is expected that the information generated by completion of Phase I will provide a valid basis for a reassessment of the balance of effort to complete the task.

ATTACHMENT 1

----- Primarily Administrative Control

——— Primarily Technical Control



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

Exhibit 1070

January 25, 1977

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

FROM: R. S. Boyd, Director, Division of Project Management

SUBJECT: DPM TASK GROUP ON POST-CP APPLICATION AMENDMENTS

In response to the directive of your January 24 memorandum, I have established a task group to compile principal architectural and engineering criteria and to develop decisional criteria to judge post-CP design changes. In addition, the group will be instrumental in evaluating the present in-house post-CP application amendments to see if any represent changes to the principal architectural and engineering criteria.

Dick DeYoung will head the group, which will include John Angelo and Bill Kane, two of our LPMs. All work of the group will be reviewed by an advisory group consisting of Dom Vassallo, Karl Kniel (two of the authors of the earlier task force report), and Larry Crocker. The work then will come to me for review, and on to Ed Case for comment.

I know you appreciate the enormity and difficulty of this task. I am not sufficiently sanguine to expect that a complete package can be developed in as short a time as two months, but we will do our best in any event.

R. S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

cc: L. V. Gossick
E. G. Case
R. E. Heineman
H. R. Denton
V. Stello
H. Shapar

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

Exhibit 1071

January 24, 1977

MEMORANDUM FOR: R. S. Boyd, Director, Division of Project Management

FROM: Ben C. Rusche, Director, Office of Nuclear Reactor Regulation

SUBJECT: POST-CP APPLICATION AMENDMENTS

The matter of handling post-CP application amendments, in light of proposed changes to principal architectural and engineering criteria, and consideration of the need for amending construction permits have been with us for sometime. Your DPM Task Force Report on Staff Review of Post-CP Design Changes is a good start on the long-term effort to develop an up-to-date scheme for processing such application amendments, and efforts to complete this work should be continued. However, it is evident that a short-term effort is required to assure that, in the interim, such matters are handled on a proper and consistent basis, and that the NRR staff understand its responsibilities in this area. To this end, I would like you to establish a task group that, over the next two months, would develop a compilation of the principal architectural and engineering criteria for the design of typical LWR plants. Recognizing the difficulty of this task, and the extremely tight schedule, the group should feel free to obtain any necessary individual consultation from other NRR staff members.

An important corollary effort to this task is for the group to develop fundamental criteria for deciding what aspects of facility design are within the confines of the principal architectural and engineering criteria. I would expect the results of this effort to be used to determine whether proposed design changes by CP licensees require a CP amendment, especially considering that even with a compilation of principal architectural and engineering criteria, ad hoc decisions will be required for many situations.

I appreciate that you have recently undertaken to have post-CP activities chronicled in the Blue Book. Necessary CP amendment decisions on each of these activities should be developed on reasonable time scales.

Please be prepared to brief me on this task group activity in about a month. Ed Case should be advised of the progress on this effort as it develops.

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

cc: L. V. Gossick
E. G. Case
R. E. Heineman
H. R. Denton
V. Stello
H. Shapar

Kl Kane

1072

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

March 7, 1977

NOTE TO: E. G. Case

POST-CP APPLICATION AMENDMENTS

Dick DeYoung, John Angelo, and Bill Kane have studied the matter of identifying the "principal architectural and engineering criteria" and have prepared the enclosed initial report. In addition to developing a scheme for formulating these criteria, they have uncovered what may be a need to regulate also on "changes to a major feature or component."

If we move from the status quo, I believe the proposals outlined in the report are as reasonable as any. What this really means is that to do anything meaningful will take a major quantum jump in how we do business. I believe, before we go much further, that the broad policy questions of this effort should be considered, and at the same time get current OELD thinking on the matter.

I suggest we discuss this with you, with a view towards briefing Ben and the NRR division directors.

Roger S. Boyd, Director
Division of Project Management

Enclosure:
As stated

cc w/o enclosure:
R. C. DeYoung
J. Angelo
W. Kane

REGULATORY GUIDE 1.XXX

PROCESSING OF LIGHT-WATER REACTOR FACILITY CHANGES SUBSEQUENT TO THE ISSUANCE OF THE CONSTRUCTION PERMIT

A. INTRODUCTION

Section 50.35, "Issuance of construction permits," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," states in part that "When an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all design features, the Commission may issue a construction permit if the Commission finds that (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design and has identified the major features or components incorporated therein for the protection of the health and safety of the public."

Section 50.34, "Contents of applications; technical information," states in part that the minimum information to be included shall consist of "an analysis and evaluation of the major structures, systems, and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in 10 C.? Part 100," and "the principal design criteria for the facility."

Condition C of each construction permit issued by the Commission states that "This construction permit authorizes the applicant to construct the facility described in the application and the hearing record, in accordance with the principal architectural and engineering criteria and environmental protection commitments set forth therein."

Regulatory Guide 1.29, "Seismic Design Classification," states in part that "Appendix A to 10 CFR Part 100 requires that all nuclear power plants be designed so that if the Safe Shutdown Earthquake occurs, all structures, systems, or components important to safety remain functional. These plant features are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100."

This guide describes an acceptable method of complying with the Commission's regulations for identification of the "principal architectural and engineering criteria" for the design of light-water reactor facilities and the "major features or components" incorporated in light-water reactor facilities for the protection of the health and safety of the public. In addition, this guide describes an acceptable method of processing post-construction permit amendments to the application as well as amendments to the construction permit.

B. DISCUSSION

1. Principal Architectural and Engineering Criteria

The NRC staff has generally held that the term "principal architectural and engineering criteria" as used in Section 50.35 of 10 CFR Part 50 has the same meaning as the term "principal design criteria" as used in Section 50.34 of 10 CFR Part 50. Section 50.34 also notes that Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 "establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provided guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units. Appendix A to 10 CFR Part 50 states in part that "the principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety." This can be assumed to define the intent of the "principal design criteria" and, therefore, the intent of the "principal architectural and engineering criteria."

On the basis of the above, it can be concluded that the "principal architectural and engineering criteria" to be identified in an application for a construction permit are to be an elaboration or amplification and extension, as necessary, of the General Design Criteria identified in Appendix A to 10 CFR Part 50. To provide a basis for consistency in applications, regulatory guidance has been developed to further define the "principal architectural and engineering criteria." Attachment A to this guide is a list of the "principal architectural and engineering criteria" which encompasses all light-water reactor facility designs. These criteria are based on the acceptance criteria provided in each section of NUREG 75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Many of the criteria specified in Attachment A are not applicable to all applications; i.e., they are for BWRs, PWRs, lake sites, river sites, etc. Therefore, each applicant for a construction permit should carefully consider each criterion of Attachment A to determine the applicability to its facility before identifying it as a "principal architectural and engineering criterion" for the facility.

2. Major Features or Components

The NRC staff has generally held that the term "major features or components" as used in Section 50.35 of 10 CFR Part 50 has the same meaning as the term "major structures, systems, and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in 10 CFR Part 100." The NRC staff has also held that the "major features or components" are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

On the basis of the above, it can be concluded that the term "major features and components" to be identified in an application for a construction permit are to be developed in accordance with the guidelines of Regulatory Guide 1.29, "Seismic Design Classification." These major features and components are to be identified in each application for a construction permit as described in Section 3.2.1 of NUREG 75/094, Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

3. Post-CP Amendments to the Application

The activities relating to post-CP amendments to facility applications, for the most part, have been confined to those amendments required to provide a basis for amending the CP. The necessity of filing such amendments is usually a result of requirements of the regulations or alterations to the terms and conditions of the CP. As a result, the changes to a facility design that routinely occur in going from a preliminary to a final design are not reflected in the Safety Analysis Report until such time as the application is amended to request an operating license (OL). At that time, the Final Safety Analysis Report (FSAR) is submitted to update the application to provide the final design information for the NRC staff's evaluation.

Although most applicants keep the NRC staff advised of such changes as they are made by means of letters, reports, meetings, etc., there is no available document in the post-CP stage that describes the current facility design. This has led to some difficulties in the basis for IE inspections which has in turn necessitated post-CP reviews of items identified by IE as possible violations of the CP. In addition, applicants have requested the NRC staff to review certain design changes to preclude potential difficulties at the OL stage of review. The NRC staff has accomplished these activities on a case-by-case basis, as necessary.

With the terms "principal architectural and engineering criteria" and "major features or components" suitably defined, it would then seem best, and in the interest of administrative consistency and convenience, and assurance of safety that all changes of significance to the proposed facility be identified and evaluated, as necessary, in amendments to the application. The changes which should require an amendment to the application for any facility licensed for construction are identified in Attachment B to this guide. These include (1) items which require an amendment to the CP and (2) items which relate to major features and in most cases would not involve an amendment to the CP as discussed in section B4 of this guide.

4. Amendments to the CP

An amendment to the CP must be applied for by the applicant in certain instances. For example, an amendment to the construction permit is required if the latest date for completion of construction, as specified in the CP, must be extended for good cause. In addition, an amendment to the CP must be sought for changes in the designation of the applicant, a serious adverse change in the applicant's financial capability, a change in the applicant's principal agents and contractors, and a change to any of the principal architectural and engineering criteria. The need for amending the CP in the event of changes of the type described above is straightforward and no special guidance appears necessary.

The additional item in Attachment B relates to changes in the "major features or components." These types of changes are more frequent in occurrence and are to be expected as the final design and construction of the facility are proceeding. These are the changes that have led and continue to lead to problems and inconsistent practices on the part of the applicants and the NRC staff. It is in this area that special guidance is needed to determine when a proposed change requires an amendment to the CP. In making such a determination, the sole test to be applied is whether implementation of such a change would alter the NRC staff's conclusion at the CP stage of review; i.e., that there is reasonable assurance such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and taking into consideration the site criteria contained in 10 CFR Part 100 the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

C. REGULATORY POSITION

1. Each application for a light water reactor facility construction permit shall identify the principal architectural and engineering criteria for the design. A composite listing of principal architectural and engineering criteria for light water reactors acceptable to the NRC staff is included as Attachment A to this guide.
2. Each application for a light water reactor facility construction permit shall identify the major features or components incorporated

therein for the protection of the health and safety of the public. The list of systems, structures, and components which meet the guidelines of Regulatory Guide 1.29 is an acceptable basis for meeting this position.

3. Amendments to the application following issuance of the CP shall be made by the applicant for each change to the facility of the type identified in Attachment B to this guide. Such amendments shall begin within one year following issuance of the CP, continue at one year intervals, and terminate one year prior to submittal of the FSAR. Each change identified as not requiring a CP amendment shall be accompanied with sufficient justification for the staff to make such an evaluation. Changes which the applicant determines require an amendment to the CP shall be processed immediately.

4. An amendment to the CP shall be required for any of the changes identified in items 1, 2, 3, and 4 of Attachment B to this guide prior to engaging in affected construction activities; i.e., the installation of affected hardware in the facility. In addition, an amendment to the CP shall be required for any of the changes to the major features or components identified in item 5 of Attachment B to this guide prior to engaging in the activities described above unless the applicant has determined that at least one of the following criteria are met:

(a) If the change was made without staff approval and was not ultimately approved by the staff at the operating license stage of review, the time required to modify the facility design to make it acceptable could be accomplished prior to the late date for completion of construction specified in the construction permit, or

(b) The change is consistent with a design or matter that was reviewed and approved by the staff in response to any of the following:

   (1) An application for a construction permit, or

   (2) An application for an operating license, or

   (3) An application for a preliminary or final design approval for a standard plant design, or

   (4) A request for review of a topical report, or

   (5) A request for review of a proposed design or matter by a licensee, or

(c) The change was identified in the Preliminary Safety Analysis Report and the staff in its Safety Evaluation Report, or supplements thereto, concluded that review of the change could be left to the operating license review stage.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for utilizing this regulatory guide.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of this guide, this guide will be used by the NRC staff on the following bases:

1. Construction permit reviews for applications docketed after ______________________, ______, will be evaluated on the basis of this guide.
2. Facilities for which construction permits have been issued prior to __________________, _____, will be evaluated on the basis of this guide, except that in position C1, the principal architectural and engineering criteria shall be defined as the general design criteria of Appendix A to 10 CFR Part 50.

ATTACHMENT A

PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA

ATTACHMENT B

CHANGES WHICH REQUIRE AN AMENDMENT TO THE APPLICATION

1. A change in the name, address or occupation or business of the licensee, or if the licensee is a partnership or corporation, a change in any of the information ini ially required by Section 50.33(d) of 10 CFR Part 50.

(2) A change in the financial condition of the licensee such that it no longer possesses the funds necessary to cover estimated construction costs and related fuel cycle costs or the assurance of obtaining the necessary funds, or a combination of the two.

(3) A change to any of the principal agents and contractors identified in the Safety Analysis Report.

(4) A change to any c: the principal architectural and engineering criteria.

(5) A change in the major features of components which requires a change in the research and development program designed to resolve safety questions associated with these features.

(6) A change to the configuration of the facility as described in the Safety Analysis Report; i.e., the addition, deletion, or relocation of any of the major features or components incorporated in the facility for the protection of the health and safety of the public.

(7) A change in the facility design involving the substitution of major feature or component A with major feature or component B where A was the major feature or component proposed in the Safety Analysis Report and the principles of operation or the type of equipment used in B represent a significant technological change from that of A.

(8) The use of design input values; e.g., loads, deflections, etc., in the final design of major features or components which are less conservative than those presented in the Safety Analysis Report.

(9) The use of analytical procedure in the final design of major features or components which are different than those presented in the Safety Analysis Report.

(10) The use of limits in the final design of major features or components which are less conservative than those presented in the Safety Analysis Report, except where the staff normally accepts less restrictive limits at the final design stage; e.g., accident doses and containment pressure.

(11) The use of codes, standards, and procedures for the design and testing of major features or components which differ from those presented in the Safety Analysis Report.

(12) The use of materials for major features of components which differ from those presented in the Safety Analysis Report.

(13) Any reduction in either the quality group, seismic category, or quality assurance classification of any of the major features or components.

(14) Changes to the quality assurance program from that presented in the Safety Analysis Report.

(15) A change in the industrial security program.

(16) A change in the site evaluation factors identified in Section 100.10 of 10 CFR Part 100.

CP AMENDMENT

1. Applicant decides to make a change to the facility design which may necessitate a CP amendment. For example, it decides to replace the existing reactor protection system; i.e., that on which the CP was based, with an advanced design. The advanced design has not yet been evaluated by the staff and none of the exceptions on pages 10 and 11 apply. The applicant, therefore, must immediately notify the staff. It is envisioned that this would be by letter.

2. The staff evaluates the information submitted by the applicant and makes one of the three conclusions:

   (1) The change is such that it can be deferred to the final design stage until a review is performed.

   (2) Additional information is required in order to determine whether the review of the change can be left until the OL review.

   (3) The change is such that it will require review and an amendment to the CP prior to implementation.

3. In the case of 2(1), no additional review is performed by the staff until the OL. However, an applicant will amend the application to describe the new design.

4. In the case of 2(2), additional information is requested and reviewed until a decision 2(1) or 2(3) can be reached.

5. In the case of 2(3), the staff will advise the applicant of its decision. At that point the applicant has the choice of:
   (1) Not proceeding with the change; i.e., using the design presented in the PSAR, or
   (2) Amending the application and undergoing the normal staff review which would be equivalent to a CP type of review.
6. If 5(2) is elected, the staff will go through the normal question and answer routine followed by issuance of a limited safety evaluation considering only that matter for which the applicant has requested an amendment of the CP. The applicant would not implement the change; i.e., in the form of installing hardware in the plant until the amendment to the CP was granted.
7. The amendment to the CP would be written to approve only those changes for which the applicant requested an amendment to the CP. Other matters which involved post-CP amendments to the application only would not be included in the amended CP unless specifically requested by the applicant.

1. SECTION 50.35(A) OF 10 CFR PART 50, ISSUANCE OF CONSTRUCTION PERMITS, STATES:

WHEN AN APPLICANT HAS NOT SUPPLIED INITIALLY ALL OF THE TECHNICAL INFORMATION REQUIRED TO COMPLETE THE APPLICATION AND SUPPORT THE ISSUANCE OF A CONSTRUCTION PERMIT WHICH APPROVES ALL PROPOSED DESIGN FEATURES, THE COMMISSION MAY ISSUE A CONSTRUCTION PERMIT IF THE COMMISSION FINDS THAT (1) THE APPLICANT HAS DESCRIBED THE PROPOSED DESIGN OF THE FACILITY, INCLUDING, BUT NOT LIMITED TO, THE PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA FOR THE DESIGN, AND HAS IDENTIFIED THE MAJOR FEATURES OR COMPONENTS INCORPORATED THEREIN FOR THE PROTECTION OF THE HEALTH AND SAFETY OF THE PUBLIC; ....

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2. SECTION 3C OF A TYPICAL CONSTRUCTION PERMIT STATES:

3. THIS PERMIT SHALL BE DEEMED TO CONTAIN AND BE SUBJECT TO THE CONDITIONS SPECIFIED IN SECTIONS 50.54 AND 50.55 OF SAID REGULATIONS: IS SUBJECT TO ALL APPLICABLE PROVISIONS OF THE ACT, AND RULES, REGULATIONS, AND ORDERS OF THE COMMISSION NOW OR HEREAFTER IN EFFECT: AND IS SUBJECT TO THE CONDITIONS SPECIFIED OR INCORPORATED BELOW:

C. THIS CONSTRUCTION PERMIT AUTHORIZES THE APPLICANT TO CONSTRUCT THE FACILITY DESCRIBED IN THE APPLICATION AND THE HEARING RECORD, IN ACCORDANCE WITH THE PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA AND ENVIRONMENTAL PROTECTION COMMITMENTS SET FORTH THEREIN.

SECTION 50.34(A)(1) OF 10 CFR PART 50, CONTENTS OF APPLICATIONS: TECHNICAL INFORMATION STATES IN PART THAT:

(A) PRELIMINARY SAFETY ANALYSIS REPORT. EACH APPLICATION FOR A CONSTRUCTION PERMIT SHALL INCLUDE A PRELIMINARY SAFETY ANALYSIS REPORT. THE MINIMUM INFORMATION TO BE INCLUDED SHALL CONSIST OF THE FOLLOWING:

(3) THE PRELIMINARY DESIGN OF THE FACILITY INCLUDING:

(i) THE PRINCIPAL DESIGN CRITERIA FOR THE FACILITY. APPENDIX A, GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS, ESTABLISHES MINIMUM REQUIREMENTS FOR THE PRINCIPAL DESIGN CRITERIA FOR WATER-COOLED NUCLEAR POWER PLANTS SIMILAR IN DESIGN AND LOCATION TO PLANTS FOR WHICH CONSTRUCTION PERMITS HAVE PREVIOUSLY BEEN ISSUED BY THE COMMISSION AND PROVIDES GUIDANCE TO APPLICANTS FOR CONSTRUCTION PERMITS IN ESTABLISHING PRINCIPAL DESIGN CRITERIA FOR OTHER TYPES OF NUCLEAR POWER UNITS:

APPENDIX A TO 10 CFR PART 50 STATE :

1. THE PRINCIPAL DESIGN CRITERIA ESTABLISHED THE NECESSARY DESIGN, FABRICATION, CONSTRUCTION, TESTING, AND PERFORMANCE REQUIREMENTS FOR STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY: THAT IS, STRUCTURES SYSTEMS AND COMPONENTS THAT PROVIDE REASONABLE ASSURANCE THAT THE FACILITY CAN BE OPERATED WITHOUT UNDUE RISK TO THE HEALTH AND SAFETY OF THE PUBLIC. THESE GENERAL DESIGN CRITERIA ESTABLISH MINIMUM REQUIREMENTS FOR THE PRINCIPAL DESIGN CRITERIA FOR WATER-COOLED NUCLEAR POWER PLANTS SIMILAR IN DESIGN AND LOCATION TO PLANTS FOR WHICH CONSTRUCTION PERMITS HAVE BEEN ISSUED BY THE COMMISSION.
2. THE DEVELOPMENT OF THESE GENERAL DESIGN CRITERIA IS NOT YET COMPLETE. FOR EXAMPLE, SOME OF THE DEFINITIONS NEED FURTHER AMPLIFICATION. ALSO, SOME OF THE SPECIFIC DESIGN REQUIREMENTS FOR STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY HAVE NOT AS YET BEEN SUITABLY DEFINED. THEIR OMISSION DOES NOT RELIEVE ANY APPLICANT FROM CONSIDERING THESE MATTERS IN THE DESIGN OF A SPECIFIC FACILITY AND SATISFYING THE NECESSARY SAFETY REQUIREMENTS.
3. IT IS EXPECTED THAT THE CRITERIA WILL BE AUGMENTED AND CHANGED FROM TIME TO TIME.............

CONCLUSIONS REGARDING PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA

1. THE TERM PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA AS USED IN THE REGULATIONS AND IN OUR CONSTRUCTION PERMITS HAS THE SAME MEANING AS THE TERM PRINCIPAL DESIGN CRITERIA AS USED IN THE REGULATIONS.
2. THE MINIMUM REQUIREMENTS FOR PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA ARE THE GENERAL DESIGN CRITERIA OF APPENDIX A TO 10 CFR PART 50.
3. THE REGULATIONS ANTICIPATED THAT AUGMENTATION OF THE GENERAL DESIGN CRITERIA WOULD BE REQUIRED.
4. THE PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA TO BE DESCRIBED IN A CONSTRUCTION PERMIT SHOULD BE AN ELABORATION OR AMPLIFICATION AND EXTENSION, AS NECESSARY OF THE GENERAL DESIGN CRITERIA.
5. REGULATORY GUIDANCE IS NEEDED TO FURTHER DEFINE THE PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA.

APPROACH USED TO ESTABLISH THE PRI CIPAL ARCHITECTURAL AND ENGINEERING CRITERIA

1. UTILIZE THE ACCEPTANCE CRITERIA OF THE STANDARD REVIEW PLAN IN ORDER TO IDENTIFY THE PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA.
2. EXPANDED CRITERIA SHOULD BE ESSENTIALLY DEVOID OF SPECIFIC NUMBERS SIMILAR TO THE GENERAL DESIGN CRITERIA.
3. USE A MINIMUM OF EDITORIAL LICENSE TO ASSURE CONSISTENCY OF TERMINOLOGY.
4. PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA HAVE BEEN DEVELOPED BY THE TASK FORCE FOR SECTIONS 2, 4, AND 5, OF THE STANDARD REVIEW PLAN.
5. WE EXPECT THAT ON THE ORDER OF 700 PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA WILL BE DEVELOPED WHEN ALL SECTIONS OF THE STANDARD REVIEW PLAN ARE INCLUDED IN THIS TABULATION.

SUGGESTED IMPLEMENTATION

1. REVISE THE STANDARD REVIEW PLAN TO INCLUDE IN SECTION 1 THE EXPANDED SET OF PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA.
2. REVISE THE STANDARD FORMAT AND CONTENT DOCUMENT TO REQUIRE AN APPLICANT TO PROVIDE IN SECTION 1.2 OF THE PSAR THE PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA BASED ON THE GUIDANCE PROVIDED IN THE STANDARD REVIEW PLAN.
3. ACCOMPLISH 1 AND 2 ABOVE ON A SCHEDULE CONSISTENT WITH THAT ESTABLISHED FOR CHANGES IN THESE DOCUMENTS AS A RESULT OF THE DIRECTOR'S MEMORANDUM OF 1/31/77 WHICH CALLS FOR IDENTIFICATION BY 5/1/77 OF MODIFICATIONS NEEDED TO THE STANDARD REVIEW PLAN TO ASSURE THAT ALL REQUIREMENTS THEREIN ARE NECESSARY, REALISTIC, AND PRACTICAL OF ACHIEVEMENT.

CRITERIA FOR DETERMINING IF A POST CP APPLICATION SHOULD BE AMENDED

1. AT PRESENT, THERE ARE NO CRITERIA FOR DETERMINING WHEN A POST-CP APPLICATION SHOULD BE AMENDED OTHER THAN THOSE CHANGES WHICH REQUIRE AN AMENDMENT TO THE CP.
2. AMENDMENTS TO THE APPLICATION POST-CP SERVE TO MAINTAIN AN ADMINISTRATIVE CONSISTENCY AS WELL AS TO ASSURE THAT SAFETY QUESTIONS ARE EVALUATED BY THE STAFF PRIOR TO CONSTRUCTION.
3. PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA ALONE ARE NOT DEFINITIVE ENOUGH TO ASSURE THAT UNREVIEWED SAFETY QUESTIONS ARE DEALT WITH PRIOR TO CONSTRUCTION.
4. WITH APPROPRIATE AMENDMENTS TO THE APPLICATION, THE PSAR CAN SERVE AS A VIABLE TOOL FOR AIDING IN I&E INSPECTION.
5. BASED ON ALL OF THESE CONSIDERATIONS, THE TASK FORCE DEVELOPED A LIST OF CRITERIA WE BELIEVE SHOULD SERVE AS THE BASIS FOR AN APPLICANT TO DETERMINE WHETHER AN AMENDMENT TO THE APPLICATION IS REQUIRED. (SEE PAGE 6 OF HANDOUT.)

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"PRINCIPAL ARCHITECTURAL & ENGINEERING CRITERIA"

EXhibit 1073

1. PRINCIPAL DESIGN CRITERIA
2. ESSENTIAL ELEMENTS OF PROPOSED DESIGN OF CERTAIN STRUCTURES, SYSTEMS & COMPONENTS
3. DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA
4. ESSENTIAL ELEMENTS OF QA PROGRAM.

(- FROM FR VOL 35, NO. 62, TUES. 3/31/70, PAGE 5317)

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tion of Orange County in Virginia because of the existence of hog cholera. This action is deemed necessary to prevent further spread of the disease. The restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will apply to the quarantined area designated herein.

The amendment imposes certain further restrictions necessary to prevent the interstate spread of hog cholera and must be made effective immediately to accomplish its purpose in the public interest. Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendment are impracticable and contrary to the public interest, and good cause is found for making it effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 25th day of March 1970.

R. J. ANDERSON,
Acting Administrator,
Agricultural Research Service.

[F.R. Doc. 70-3796; Filed, Mar. 30, 1970; 8:46 a.m.]

Title 10—ATOMIC ENERGY

Chapter I—Atomic Energy Commission

BACKFITTING OF PRODUCTION AND UTILIZATION FACILITIES; CONSTRUCTION PERMITS AND OPERATING LICENSES

On April 16, 1969, the Atomic Energy Commission published for comment in the FEDERAL REGISTER proposed amendments and the proposed definition in § 50.2 of the "principal architectural and engineering criteria" of the proposed design of a facility; (2) the addition of conforming amendments to Part 170; and (3) the addition of minor corrective amendments to §§ 50.35, 50.57 and proposed § 50.109.

The rapid changes in technology in the field of atomic energy result in the continual development of new or improved features designed to improve the safety of production and utilization facilities. Section 50.109 which follows defines the circumstances under which the Commission may require backfitting of facilities—that is, the addition or modification of structures, systems or components affecting the safety of the facility after the construction permit has been issued. It provides that the Commission may require backfitting if it finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security.

Section 50.109 is not, however, intended to affect the responsibility of applicants for, or holders of, facility licenses to evaluate significant new information developed as a result of experience in the design, construction, testing and operation of facilities and the results of research and development programs bearing on the safety of facilities, and to recommend any additions to, or modification of facilities needed to protect the health and safety of the public.

In the past the Commission has issued "provisional" construction permits when an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all proposed design features. In practice, almost all construction permits have never been converted into "final" construction permits, but have merged directly into the operating li-

units referred to in the proposed rule require further definition involving additional study. Accordingly, the proposed amendments of § 50.35 other than those eliminating the term "provisional" construction permit and a related note and the proposed definition of "principal architectural and engineering criteria" in § 50.2 have not been adopted at this time.

By amendments to § 50.57, the "provisional" operating license, which is issued for an 18-month period, is eliminated. Temporary limitations on operation considered necessary for public health and safety will be incorporated in the full-term operating license as conditions. The elimination of the provisional operating license does not preclude the Commission from imposing all the limitations in the full-term operating license which may have been required in the provisional operating license. The findings required for issuance of an operating license are largely the same as those which have been required for a provisional operating license. The elimination of the provisional operating license removes one step in AEC's facility licensing process and is expected to reduce the time consumed in the facility licensing process without reducing the degree of protection of the public health and safety provided. Provisional operating licenses already issued will continue in effect in accordance with their terms. Conforming amendments with respect to the operating license have also been made to Parts 2 and 170.

Pursuant to the Atomic Energy Act of 1954 as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter 1, Code of Federal Regulations, Parts 2, 50 and 170, are published as a document subject to codification to be effective 30 days after publication in the FEDERAL REGISTER.

westerly direction to Secondary Highway 621; thence, following Secondary Highway 621 in a generally southwesterly direction to Secondary Highway 608; thence, following Secondary Highway 608 in a generally southerly direction to the Orange-Spotsylvania County line; thence, following the Orange-Spotsylvania County line in a southwesterly direction to Secondary Highway 651; thence, following Secondary Highway 651 in a generally southwesterly direction to Secondary Highway 629; thence, following Secondary Highway 629 in a generally northwesterly direction to U.S. Highway 522; thence, following U.S. Highway 522 in a generally northerly direction to its junction with Secondary Highway 663.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended)

Effective date. The foregoing amendment shall become effective upon issuance.

The amendment quarantines a portion of Orange County in Virginia because of the existence of hog cholera. This action is deemed necessary to prevent further spread of the disease. The restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will apply to the quarantined area designated herein.

The amendment imposes certain further restrictions necessary to prevent the interstate spread of hog cholera and must be made effective immediately to accomplish its purpose in the public interest. Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendment are impracticable and contrary to the public interest, and good cause is found for making it effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 25th day of March 1970.

R. J. ANDERSON,
Acting Administrator, Agricultural Research Service.

[F.R. Doc. 70-3796; Filed, Mar. 30, 1970; 8:46 a.m.]

Title 10—ATOMIC ENERGY

Chapter I—Atomic Energy Commission

BACKFITTING OF PRODUCTION AND UTILIZATION FACILITIES; CONSTRUCTION PERMITS AND OPERATING LICENSES

On April 16, 1969, the Atomic Energy Commission published for comment in the FEDERAL REGISTER proposed amendments to its Rules of Practice, 10 CFR Part 2, and to its regulation, Licensing of Production and Utilization Facilities, 10 CFR Part 50, which would (1) define more precisely the significance of the issuance of a construction permit for a facility, (2) simplify and expedite the Commission's facility licensing process by eliminating the "provisional" operating license, and (3) clarify the Commission's position with respect to requirements for additional safety features after the issuance of a construction permit (34 F.R. 6540).

All interested persons were invited to submit written comments and suggestions for consideration in connection with the proposed amendments within 60 days after publication of the notice of proposed rule making in the FEDERAL REGISTER. Upon consideration of the comments received and other factors involved, the Commission has adopted the amendments set out below. The amendments are the same as the proposed amendments published April 16, 1969, except for (1) the elimination of the proposed amendments to § 50.35, other than those deleting the term "provisional" construction permit and a related note, and the proposed definition in § 50.2 of the "principal architectural and engineering criteria" of the proposed design of a facility; (2) the addition of conforming amendments to Part 170; and (3) the addition of minor corrective amendments to §§ 50.35, 50.57 and proposed § 50.109.

The rapid changes in technology in the field of atomic energy result in the continual development of new or improved features designed to improve the safety of production and utilization facilities. Section 50.109 which follows defines the circumstances under which the Commission may require backfitting of facilities—that is, the addition or modification of structures, systems or components affecting the safety of the facility after the construction permit has been issued. It provides that the Commission may require backfitting if it finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security.

Section 50.109 is not, however, intended to affect the responsibility of applicants for, or holders of, facility licenses to evaluate significant new information developed as a result of experience in the design, construction, testing and operation of facilities and the results of research and development programs bearing on the safety of facilities, and to recommend any additions to, or modification of facilities needed to protect the health and safety of the public.

In the past the Commission has issued "provisional" construction permits when an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all proposed design features. In practice, almost all construction permits have never been converted into "final" construction permits, but have merged directly into the operating license. The amendments of § 50.35 and conforming amendments to Parts 2 and 170 which follow eliminate the term "provisional" construction permit, thus conforming the terminology with Commission practice. The findings required for issuance of a construction permit would be the same as those which have been required for a "provisional" construction permit.

The proposed amendment to § 50.35 would have provided that the Commission, in issuing a construction permit, would be approving the construction of the facility in accordance with the application, including the principal architectural and engineering criteria. "Principal architectural and engineering criteria" would have been defined, by amendment of § 50.2, to include (1) the principal design criteria, (2) the essential elements of the proposed design for certain structures, systems and components, (3) the design bases for protection against natural phenomena, and (4) the essential elements of the applicant's quality assurance program. On further consideration, it appears that the "essential elements of the proposed design" of the structures, systems and components of water-cooled nuclear power units referred to in the proposed rule require further definition involving additional study. Accordingly, the proposed amendments of § 50.35 other than those eliminating the term "provisional" construction permit and a related note and the proposed definition of "principal architectural and engineering criteria" in § 50.2 have not been adopted at this time.

By amendments to § 50.57, the "provisional" operating license, which is issued for an 18-month period, is eliminated. Temporary limitations on operation considered necessary for public health and safety will be incorporated in the full-term operating license as conditions. The elimination of the provisional operating license does not preclude the Commission from imposing all the limitations in the full-term operating license which may have been required in the provisional operating license. The findings required for issuance of an operating license are largely the same as those which have been required for a provisional operating license. The elimination of the provisional operating license removes one step in AEC's facility licensing process and is expected to reduce the time consumed in the facility licensing process without reducing the degree of protection of the public health and safety provided. Provisional operating licenses already issued will continue in effect in accordance with their terms. Conforming amendments with respect to the operating license have also been made to Parts 2 and 170.

Pursuant to the Atomic Energy Act of 1954 as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter 1, Code of Federal Regulations, Parts 2, 50 and 170, are published as a document subject to codification to be effective 30 days after publication in the FEDERAL REGISTER.

PART 2—RULES OF PRACTICE

§ 2.104 [Amended]

1. Section 2.104(b)(2) and sections I (c) and (d), III(g)(1) and IV (c) and (d) of Appendix A of 10 CFR Part 2 are amended by substituting the words "construction permit" for "provisional construction permit" where they appear.

§ 2.761 [Amended]

2. Paragraph (d) of § 2.761 of 10 CFR Part 2 is amended by substituting the words "operating license or provisional operating authorization" for "provisional operating license or authorization".

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

3. Paragraphs (a) and (b) of § 50.35 of 10 CFR Part 50 are amended to read as follows:

§ 50.35 Issuance of construction permits.

(a) When an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all proposed design features, the Commission may issue a construction permit if the Commission finds that (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public; (2) such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report; (3) safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and that (4) on the basis of the foregoing, there is reasonable assurance that, (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in Part 100 of this chapter, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

NOTE: When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the facility, the findings required above will be appropriately modified to reflect that fact.

(b) A construction permit will constitute an authorization to the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit. The applicant, at his option, may request such approvals in the construction permit or, from time to time, by amendment of his construction permit. The Commission may, in its discretion, incorporate in any construction permit provisions requiring the applicant to furnish periodic reports of the progress and results of research and development programs designed to resolve safety questions.

4. Section 50.57 of 10 CFR Part 50 is revised to read as follows:

§ 50.57 Issuance of operating license.

(a) Pursuant to § 50.56, an operating license may be issued by the Commission, up to the full term authorized by § 50.51, upon finding that:

(1) Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter; and

(4) The applicant is technically and financially qualified to engage in the activities authorized by the operating license in accordance with the regulations in this chapter; and

(5) The applicable provisions of Part 140 of this chapter have been satisfied; and

(6) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

(b) Each operating license will include appropriate provisions with respect to any uncompleted items of construction and such limitations or conditions as are required to assure that operation during the period of the completion of such items will not endanger public health and safety.

(c) In a case where a hearing has been held in connection with a proceeding under this section the presiding officer may, upon written motion and upon good cause shown, provide that any initial decision issued pursuant to this section shall become effective ten (10) days after issuance subject to (1) the review thereof and further decision by the Commission or the Atomic Safety and Licensing Appeal Board, as appropriate, upon exceptions filed by any party, and (2) such order as the Commission or the Atomic Safety and Licensing Appeal Board may enter upon such exceptions or upon its own motion within forty-five (45) days after the issuance of such initial decision. In the absence of a Commission or an Appeal Board order pursuant to the foregoing, and in the absence of exceptions to the initial decision, the initial decision shall become the final decision of the Commission at the end of such forty-five (45) day period. If any party opposes the motion for expedited effectiveness of the initial decision, the presiding officer may stay its effectiveness pending filing within five (5) days after its issuance of an exception to the provision for expedited effectiveness, and thereafter until decision by the Commission or the Atomic Safety and Licensing Appeal Board on the exception.

5. An undesignated center head and a new § 50.109 are added to 10 CFR Part 50 to read as follows:

BACKFITTING

§ 50.109 Backfitting.

(a) The Commission may, in accordance with the procedures specified in this chapter, require the backfitting of a facility if it finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security. As used in this section, "backfitting" of a production or utilization facility means the addition, elimination or modification of structures, systems or components of the facility after the construction permit has been issued.

(b) Nothing in this section shall be deemed to relieve a holder of a construction permit or a license from compliance with the rules, regulations, or orders of the Commission.

(c) The Commission may at any time require a holder of a construction permit or a license to submit such information concerning the addition or proposed addition, the elimination or proposed elimination, or the modification or proposed modification of structures, systems or components of a facility as it deems appropriate.

PART 170—FEES FOR FACILITIES AND MATERIALS LICENSES UNDER THE ATOMIC ENERGY ACT OF 1954 AS AMENDED

§ 170.12 [Amended]

6. Section 170.12(b) and (c) of 10 CFR Part 170 are amended by substituting the words "construction permit" for "provisional construction permit," and "operating license" for "provisional operating license" where they appear.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at Washington, D.C., this 20th day of March 1970.

For the Atomic Energy Commission.

P. T. HOBBS,
Acting Secretary.

[F.R. Doc. 70-3799; Filed, Mar. 30, 1970; 8:46 a.m.]

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

Exhibit
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December 23, 1975

Roger S. Boyd, Deputy Director, Division of Reactor Licensing

TASK FORCE REPORT ON STAFF REVIEW OF POST-CP DESIGN CHANGES

Enclosed are ten copies of the Task Force Report on Staff Review of Post-CP Design Changes for distribution as you desire. Members of the Task Force would be interested in participating in any additional efforts that might be made to establish the feasibility of the recommendations made in the report.

Karl Kniel

Karl Kniel, Chairman
Task Force

cc:
D. Vassallo
K. Kniel

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December 1975

TASK FORCE REPORT

ON

STAFF REVIEW OF POST-CP DESIGN CHANGES

Office of Nuclear Reactor Regulation
Division of Reactor Licensing

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1. <u>SUMMARY</u>

The task force was directed to review our current practice of handling design changes during the post CP-review phase and to recommend changes, either by procedure, rule, or legislation, which would bind permit holders to the representations made ~~in~~ their in their PSAR and the hearing record. The approach selected was to be practicable to both applicants and NRC staff, wihtout the necessity of incorporating the entire verbatim PSAR. Any proposals should also recognize that the design is "preliminary" and sufficient flexibility should be afforded to accommodate needed, as well as desirable, design changes evolving during the plant construction phase.

We found that past practice, both from the licensee's viewpoint, as well as from the staff's, has varied. The Task Force distilled the basic requirements of the study into the following:

(1) To provide a licensee with guidance and a legal basis for requiring staff review of changes to PSAR representations following issuance of a construction permit.

(2) To provide a definitive legal basis for acceptance by the staff of the facility design approved for the construction permit so as to limit staff racheting at the operating licensing stage.

(3) To provide the Office of Inspection and Enforcement with the basis for conducting and applying a more objective and consistent means of inspecting and enforcing the facility design approved in a construction permit during the construction phase.

The most feasible approach appears to be that of developing a document defining the "Design Features" of the plant being approved. This document would emphasize the essential design criteria used as a basis for NRC approval largely by reference to existing criteria, guides, and standards and would include only enough descriptive material on sites, architectural and engineering arrangements, and procedures to permit the criteria to be readily comprehensible.

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Appropriate rule changes are proposed which would require construction permit holders to obtain prior approval in the form of a CP amendment from NRC. A corollary change would be required in the "regulatory philosophy" whereby the emphasis during the OL review would be on design implementation and verification of technical specifications. Plant design would be approved and "fixed" at the point in time at which the CP was issued unless the "backfit" rule (substantial, additional protection which is required for the public health and safety) could be justified or new requirements are issued as regulations.

2. INTRODUCTION

Neither the Atomic Energy Act nor the NRC's regulations precisely define the legal commitment binding upon an applicant upon the granting of a construction permit for a nuclear power plant. As a result, there are conflicting opinions concerning a construction permit's legal effect, particularly with regard to whether a utility is bound by representations made in its application, particularly in the PSAR, or on the hearing record.

For years there has been concern as to the appropriate procedures and licensing actions that should be taken regarding design and other changes to a nuclear power plant after a construction permit has been issued and prior to issuance of an operating license. Past practice both by holders of construction permits and the NRC staff has been varied, with changes being implemented on an "ad hoc" basis. The mechanism used by permittees to inform the NRC of changes to the PSAR have been varied. These range from formal letters and amendments, to submittal of informal drafts and oral communication, or to making no notification of changes until submittal of the FSAR. The staff's responses to these actions have ranged from preparing a written safety evaluation, to use of letters acknowledging a change and notifying the permittee that the matter would be reviewed in the OL stage, oral acknowledgement, or a review and decision to defer action to some later date. Accordingly, it is apparent that the NRC has no clear, consistent position on the legal enforcement status of a construction permit and the circumstances which warrant an amendment.

The problem presented to the task force then, was to recommend procedures, criteria, and/or changes to regulations which would bind more precisely permit holders to the representations in their PSARs and the hearing record.

Although not expressly directed to do so, the task force felt that an equally important objective of the study was to consider the attendant obligations of the NRC staff associated with any proposed changes in methods for licensing actions in the interim period between issuance of a construction permit to the submittal of an application for an operating license.

3. DISCUSSION

The Task Force reviewed the historical record of the types of changes that have been made by permit holders and the staff's bases for review of such changes. It found that the staff has not provided much guidance to permit holders on how to handle changes to the PSAR. No written categories of types of proposed changes which would require the permittee to take some particular kind of action, such as filing an amendment to the CP or to the PSAR, are available. In the past, the staff has reacted to proposed changes on a case-by-case basis. Licensing history is replete with the different types of action taken by the permittees and the staff. The following are a few examples.

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(a) Applicant may choose not to inform the staff. For example, in Indian Point 2, Consolidated Edison did not inform the staff until the time of the FSAR that it was not constructing a "core catcher" as was described in the PSAR.

(b) Applicant may informally inform the staff. For example, in San Onofre, Units 2 and 3, Southern California Edison informed the staff at a meeting on April 23 and 25, 1974, that it would replace the original proposed 14 x 14 core with the new Combustion Engineering 16 x 16 core.

(c) Applicant may formally inform the staff of the change. For example, in Summer, Unit 1, South Carolina Electric & Gas Company, in a letter to the staff, described a design modification using the Westinghouse 17 x 17 core design.

(4) Formal submittal by applicant and approval by NRC. Georgia Power Company submitted an application amendment to permit joint participation in the ownership (30% undivided interest) by Oglethorpe Electric Membership Corporation. The staff completed its review, prepared a safety evaluation, prepared a Federal Register Notice, and issued a CP amendment.

This case is covered elsewhere in the regulations.

As previously stated, the staff's responses have varied. Most frequently the staff has taken the position that any change proposed by a permittee following CP issuance will be reviewed in detail at the OL stage. Thus, the theory is that any proposed change will be made at the permit holder's own risk. However, if the staff considers a proposed change significant and judges that the matter must be resolved before construction of the facility proceeds too far, then the staff has in the past initiated a review. In some cases, this has been followed with a formal letter to the permittee stating the staff's views concerning the proposed change.

An associated problem with the OL review is the staff's propensity to require a facility design, ostensibly approved at the CP stage, to be updated to meet current requirements. The term "ratcheting" has been coined to describe this type of practice, although the practice is not new. It is the Task Force's opinion that, if new methods are established to bind a permittee to the representations in its PSAR, then it only seems logical that the staff must itself develop a disciplined approach to requiring changes after the issuance of a CP.

The ratcheting-at-the OL-review approach developed over a period of time for several reasons. Basically, it appears that this difficulty arose because the Commission's regulations do not expressly prohibit changes in a facility design after a CP is issued. Accordingly, the licensee could and did make changes at its own discretion. In these instances, at least in the past, the licensee did not always make a comprehensive listing in the FSAR of all design changes made after issuance of the CP. Therefore the staff developed a philosophy that it had no recourse but to review the entire application at the OL stage to first determine the adequacy of the basic design and then to assess the implementation. In reality then, this practice has resulted in the staff conducting another CP review at the OL stage and in the process of re-reviewing the basic design the staff has frequently required updating of the application to meet new staff requirements. This practice can be interpreted to be condoned by the regulations inasmuch as 50.35(b) states:

> "A construction permit will constitute an authorization to the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit...." (emphasis supplied).

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and 50.35(c) states:

> "Any construction permit will be subject to the limitation that a license authorizing operation of the facility will not be issued by the Commission until .... and (2) the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility...."

Therefore, in order to control ratcheting, the Task Force believes that the staff must also be bound to accepting an approved PSAR and to require changes through a more formalized procedure such as the use of 10 CFR 50.109 (Backfitting) or a variation of it.

Another facet to the problem of design changes following issuance of a CP is that I&E inspects the proposed facility under construction to determine if the applicant is complying with the representations made in the PSAR and the hearing record. When I&E feels that an applicant has made some change from that described in the PSAR, then I&E tries to assure that the applicant has or will inform Licensing of this change. The actions taken by I&E have also been somewhat inconsistent. Sometimes, because an applicant feels that he may be faced with a citation from I&E, the applicant has sought approval from Licensing for changes of representations made in the PSAR. In other cases, I&E has or has tried to cite an applicant for a violation for even minor changes from that described in the PSAR. In some cases, I&E checks with Licensing to resolve a potential problem concerning an applicant's proposed change. Other times I&E leaves this up to the applicant to do.

Because there is considerable doubt concerning a construction permit's legal status, it appears that I&E has found it difficult to provide consistent guidance in conducting inspections during the construction phase of a facility. Therefore, any new method for binding an applicant to specific representations as a result of the issuance of a construction permit must consider inspection and enforcement functions.

After a review of past practices regarding licensing actions following issuance of a construction permit, the Task Force believes that three basic requirements should be considered in proposing new methods for binding holders of construction permits to representations of a facility design in the PSARs. These are:

(1) To provide a licensee with guidance and a legal basis for requiring staff review of changes to PSAR representations following issuance of a construction permit.

(2) To provide a definitive legal basis for acceptance by the staff of the facility design approved for the construction permit so as to limit staff racheting at the operating license stage.

(3) To provide the Office of Inspection and Enforcement with the basis for conducting and applying a more objective and consistent means of inspecting and enforcing during construction of the facility design approved in a construction permit.

A change to the current policy for handling post-CP reviews and ratcheting of OL reviews might be accomplished in one of three ways: (1) through revised internal administrative procedures; (2) changes in the regulations; and/or (3) changes in the authorizing legislation.

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Alternative 1, the institution of internal administrative procedures, does not appear to be an effective mechanism for accomplishing the task. We presently have ad hoc procedures ennunciated in the Project Managers Handbook as well as the functioning of the Regulatory Requirements Review Committee. These have not been sufficiently adequate tools to enable the staff to require applicants to file post-CP submittals for review in an orderly fashion. Likewise, the staff, both technical and legal, do not have bases for approving or disapproving design changes occurring after CP issuance and prior to OL issuance, unless such action is specifically requested by an applicant. While management guidance and direction in the form of internal procedures may be necessary to implement any rule or legislative changes, that alone will not be a strong enough requirement to provide the necessary underlying authority to the staff to assure that the plant is constructed in accordance with the principal architectural and engineering criteria which are approved during the review of the Preliminary Safety Analysis Report and as modified by the hearing record.

The second alternative of proposed rule changes, would, if effected, lend more substance to our CP findings. Presently, 10 CFR 50.36 requires each applicant "for a license authorizing operation" to include proposed Technical Specifications in the Final Safety Analysis Report. Among the items required by 10 CFR 50.36 to be included in Technical Specifications are Design Features which are defined as "those features of the facility such as materials of construction and geometric arrangements, which if altered or modified, would have a significant effect on safety...." Since the term "principal architectural and engineering criteria" has never been defined through legislation or the regulations, it seems reasonable to conclude that indeed that's what design features relate to. Ten CFR 50.55 (e) (1) recognizes that certain deficiencies or deviations might occur in the design during the construction process to warrant reporting and other further staff review. Proposed rule changes, as discussed in the conclusions of this report, to 10 CFR 50.36, 50.55 (e), and 50.53 (b) would shift the review and approval of design features to the construction permit review stage and thus assure that the design bases set forth in the Preliminary Safety Analysis Report and approved during the CP review can be subsequently met through design features unique to that plant which are necessary to assure public health and safety.

Alternative 3, legislative change, is not necessary since Section 182 a. of the Atomic Energy Act of 1954 provides authority for the Commission to include Technical Specifications, of which design features is one segment, in licenses. Further Section 185 of the Act recognizes that this license authority may be granted in two stages: namely, construction permit and operating license.

Therefore, there is no limitation that would prohibit approving a portion of the Technical Specifications during the first, or CP, stage. New legislation proposed to the Congress this year would make this a moot point since a combined CP and OL could be issued initially when sufficient final design information is available. It would, however, be even more important to incorporate into a CP those specific design features which could not be changed.

4. PROPOSED CHANGE

The proposal is to generate a document which might be called the "Principal Architectural and Engineering Criteria" which would be the essential results of the CP review. Rather than continuing to use this old terminology, the Task Force is proposing to use the term "Design Features," which is currently part of the Technical Specifications issued with the operating license. Our proposal is to make the Design Features section of the Technical Specifications a legal part of the CP, in the same manner that is done to the entire Technical Specifications with an OL. This is not a new idea, but rather an idea whose time has come since we now have the resources in terms of reference documents which will allow such a document to adequately describe the CP commitment largely by reference and with a minimum of words and tables of its own. The principal reference documents would be the Standard Review Plan, the General Design Criteria, Regulatory Guides, Branch Technical Positions, and Industry criteria, codes, and standards to the extent necessary.

The principal purpose of such a document would be to satisfy the three requirements discussed above; namely, to serve:

(1) as a basis for an applicant decision regarding the need to request an amendment to the CP;

(2) as a basis for staff approval at the CP stage in order to serve to minimize racheting at the OL without the staff making the appropriate backfit finding required by 50.109;

(3) as a basis for I&E actions subsequent to the CP.

The function of the Technical Specifications issued with the operating license is to maintain the safety designed into the plant during operation over its operating lifetime. In a similar way, the function of the design features issued with the CP would be to assure that the safety of the conceptual plant approved during the CP in terms of design criteria is maintained in its implementation through the detailed design and construction stages.

At the present time, Section 5 of the OL Technical Specifications, which is labelled "Design Features," is a virtually totally emasculated section representing the vestigial remains of what previous to 1966 had been the entire FSAR. The revised Section 5.0 that we propose would provide a concise summary of the bases for plant design and its vintage and would also probably prove valuable to the staff after the plant goes into operation when considering the need to backfit operating plants to meet a new requirement.

Following CP issuance, the permit holder could request a change in the Design Features by submitting an application for amendment to the Construction Permit. Following staff review and appropriate findings, a CP amendment would be issued. The extent of public participation in the review of such an amendment has been considered by the Task Force. It appears that in principle, a provision for public recourse by providing for an opportunity for a public hearing is necessary. However, in order to limit hearings to the matters of real public concern and to minimize the potential for delay in the construction of the plant, the offer of opportunity for a hearing should be restricted to those changes deemed to be of sufficient significance.

At the present time, operating license amendments are pre-noticed with an opportunity for a hearing offered only in those cases where the staff makes a finding that significant hazards considerations are involved. Since a CP amendment does not have the immediacy

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of impact that an OL amendment does since the subject of a CP amendment is subject to the offer of public review at the operating license stage, it seems appropriate that CP amendments be handled somewhat differently. Consequently the Task Force suggests that the offer of opportunity for a public hearing be predicated on a finding by the staff that the proposed amendment could result in a substantial reduction in the protection which is required for the public health and safety.

It is expected that most amendments would involve changes or adjustments in design criteria resulting in approximately equivalent safety and no hearing would therefore be required. If applicants attempted to substantially reduce the commitment made to safety during the CP review and hearing, a subsequent hearing would be required.

As a step in implementing and confirming the feasibility of this concept, each Technical Review Branch would be asked to complete the requirements for its portion of the Design Features using its portion of the Standard Review Plan (SRP) as a basis. Many sections of the SRP which address systems already include reference to the Regulations, GDC, Guides, and Industry Codes and Standards. In this effort, it would be important to limit the content to essentials and address criteria and not design implementation and methods of analysis. The acceptable criteria should adequately reflect the current technology of methods of analysis and design implementations without reference to specifics in these areas.

Although it is expected that the document would appropriately serve Inspection and Enforcement and NRR in assuring that applicants implement the essential features of the CP review, the main impact would probably be a limitation on staff racheting during the OL review. The emphasis in the OL review would hopefully be shifted somewhat toward its true function of review of design implementation rather than a redone CP with emphasis on racheted criteria and resulting

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minimal time to perform a review of the design and necessary operating Technical Specifications as is presently the case. Plants would be vintaged to criteria, and visibility and recognition would be given to the passage of time and the stepwise upgrading of safety design technology with the associated distinction between desirable and necessary (i.e., backfit in the 50.109 sense) change.

Since the Design Feature document would achieve definition mostly by means of reference to other criteria, codes, guides, and the Standard Review Plan, the trend toward a more systematic technical review now underway would be strongly encouraged and staff requirements would have better visibility and be better understood by applicant, the staff, and the public.

5. RECOMMENDATIONS AND CONCLUSIONS

To accomplish the desired objective, the following should be done:

(1) If DRL management agrees that the proposal is an acceptable solution, additional sections of the Design Features should be drafted by Reactor Licensing and proposed to Technical Review for comment. (To speed up the process, individual LPMs could be assigned to prepare separate section(s).)

(2) TR, I&E, ELD comments and concurrence would be solicited.

(3) In this connection, ELD would need to prepare appropriate rule changes. The Task Force suggests that proposed rule changes be made to 10 CFR 50.36(c)(4), 50.55(3), and 50.58(b) as follows:

A. 50.36(c)(4) Design Features. Design features to be included are those features of the facility such as materials of construction and geometric arrangements, site parameters affecting design bases, provisions relating to organization and management responsible for construction of the facility, in accordance with codes, standard, and regulatory guides in effect at the time a construction permit is issued, which if altered or modified, would have a significant effect on safety and are not covered in categories described in subparagraphs (1), (2), and (3) of this paragraph (c).

B. 50.55(e) (5). If the permit is for construction of a nuclear power plant, the holder of the permit shall submit to the Commission information on any changes in the design features as described in 50.36(c)(4), which affect safety of the plant. Those proposed changes which are initiated by the licensee shall be submitted for prior review and approval. Those occurrences which are beyond the control of the licensee must be reported as soon as practicable following the licensee's knowledge of the occurrence, and shall include a description of the occurrence and a safety analysis. Specifically, changes in the area identified below (or in a new Appendix to 10 CFR Part 50) should be reported. Changes should be submitted as an amendment to the application and should provide comparative information on the nature of the change with the appropriate section of the Preliminary Safety Analysis Report, as previously amended.

C. The last sentence of 50.58(b) should be changed and added to as follows:

If the Commission finds that no significant hazards consideration is presented by an application for an amendment to an operating license, it may dispense with such notice and publication and may issue the amendment. If the Commission finds that no substantial reduction in the protection which is required for the public health and safety is presented by an application for an amendment to a construction permit, it may dispense with such notice and publication and may issue the amendment.

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(4) NRR management should issue staff guidance which clearly delineates the fact that this is a departure from the current practice of conducting a CP review prior to CP issuance, and another CP review at the OL stage, and that the scope of the review at the OL stage will be limited to implementation of the design as approved at CP issuance and those changes required by the backfit rule or change in the regulations.

Karl Kniel, Chairman

Domenic B. Vassallo

Sybil M. Kari*

Attachments:
1. 2.0 Site Characteristics
2. 6.2.1 Containment Functional Design (Dry Containment)
3. 13.1 Management and Organization

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*Sybil M. Kari is a principal contributor to the report and was a member of the Task Force prior to her resignation from NRC.

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

Exhibit
1075

January 7, 1976

Ben C. Rusche, Director, Nuclear Reactor Regulation

PROPOSED METHOD OF HANDLING POST-CP DESIGN CHANGES

For many years we have been faced with the problem of not having any real way to regulate facility design changes between the time a CP is issued and an FSAR is filed. This has led to two major problems; considerable ratcheting, whether real or imagined, controlled or uncontrolled, at the OL stage, and no clear basis by which I&E can enforce the requirements of a construction permit.

An abortive attempt to solve these problems was made in 1969 with a proposed rule that would have required the specification of the "essential elements of design" (and the essential elements of the QA program) which could not be changed without prior Commission approval. That part of the proposed rule was never adopted, principally because the idea was ahead of its time. (Interestingly, two other parts of the rule were adopted; that relating to the abolition of provisional licenses, and the backfit rule now known as 10 CFR 50.109.)

About a year and a half ago, in response to one of the recommendations of the AEC's Action Plan (authored by John Peters) and I&E's continuing request for such action, I tried to rejuvenate the concept of the "essential elements". OELD did a general study for us at that time on the legal options open towards a scheme of regulatory action down this path. With all this background information in hand I appointed a task force (with Karl Kniel as chairman) to develop a proposal for a workable plan from which the appropriate regulations and implementing procedures could be developed that would provide a specific basis for handling these design changes. The task force effort is completed, and their report is attached for your consideration.

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(Another significant piece of information in this scenario is that last November, Joe Gallo advised us that under the present rules a CP licensee, planning changes to the "principal architectural and design criteria", however ill-defined, is required to obtain an amendment to the CP. He further advised that it is the staff's responsibility, in the interim between issuance of a CP and the application for an operating license, to assure adherence to those criteria. Joe, in my opinion, is right, but we presently have no systematic way of doing this.)

The system we need, and the one the task force endeavored to find, is one that would provide clear specificity on what a licensee could and could not change; something that would be "enforceable", and a new mode of doing business that would not be unduly burdensome or otherwise contribute to delays, either on us or the licensees. I think the proposed system has the potential for meeting these goals.

Essentially, the task force proposes a system whereby the "design features" of a plant would be specified. The idea is not new; what is new is the analysis contained in their report to persuade people that this is the most meaningful solution to the problem. In addition, they have developed an example of part of such a "design features" document as an example to aid in reaching a management decision on this proposal.

I think the NRR senior staff should get together to consider this policy question. If we agree with the task force proposal, I believe the report is in a form suitable for a Commission paper, with a suitable cover report memo that could be developed easily.

Roger S. Boyd

Roger S. Boyd, Acting Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachments:
1. Memo from K. Kniel
   dtd Dec. 23, 1975
2. Task Force Report

cc: E. G. Case
R. E. Heineman
H. R. Denton
V. Stello
R. C. DeYoung
K. Kniel

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

DeYoung

June 27, 1976

Exhibit 1076

Roger S. Boyd, Director, Division of Project Management
Harold R. Denton, Director, Division of Site Safety and Environmental Analysis
Robert E. Heineman, Director, Division of Systems Safety
Victor Stello, Director, Division of Operating Reactors

POST CP DESIGN CHANGES

For some time, we have not had a consistent procedure for handling nuclear power plant design changes subsequent to issuance of a construction permit. As a result of this, a task force was formed and reported to R. S. Boyd on December 23, 1975 with several recommendations. A copy of that report is enclosed. The task force proposed that a set of design features be included with the CP. This package would be to the CP what the Tech Specs are to the OL and would serve to provide IE with a specific description of the basic design as well as restrict ratcheting at the OL phase of review. These design features could not be changed without prior Commission approval.

I would like to discuss this proposal with you in about two weeks, after you have had time for staff review of the proposal. Several issues should be included during this review and subsequent discussion. They include 1) the effects of such a requirement on Quality Assurance activities, 2) whether certain of the proposed items would be actually under the control of the licensee and whether they could be monitored by IE (e.g. population within 10 miles of the site), 3) the level of detail that should be specified, 4) the usefulness of this approach during the FSAR review, 5) the increase in workload expected to result from such a requirement, 6) the type of change that could be permitted without prior notification and Commission approval, and 7) the method for allowing public participation during consideration of the change.

Ben C. Rusche

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

cc: L. V. Gossick

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Brenda - please check this one again —
Thanks
Jim Long
(28041)

UNION OF CONCERNED SCIENTISTS

January 6, 1977
Exhibit 1077

J. M. Felton, Director
Division of Rules and Records
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Felton:

Re: FOIA-76-397

In your letter of December 21, 1976, you stated that "we could find no record of item 2, a letter dated September 8, 1975 to Mississippi Power and Light Company from W. R. Butler regarding Grand Gulf." I am writing, as you suggested during a recent telephone conversation, to provide additional information to assist your locating the document.

The document I requested is in the form of a letter to Mississippi Power and Light Company from W. R. Butler of the NRC staff. The subject matter of the document relates to the staff's review of Amendment 22 to the PSAR for Grand Gulf and the staff's determinations that, since a "proposed" change is a change to the principal architectural and engineering criteria, application for amendment of the construction permits should be filed. According to the yellow (concurrence) file copy, the document was approved by E. H. Butcher, the Site Analysis Branch and the Office of the Executive Legal Director. The document was approved and signed by W.R. Butler on or about September 8, 1975.

I suggest that you contact Messrs. Butcher and Butler and the individuals in the SAB and OELD assigned to Grand Gulf at that time in order to locate the requested document. Since you did not respond to my FOIA request within the time limits prescribed by the law and since I have been assured that the requested document exists, I expect that you will respond to this letter by promptly producing the requested document.

Sincerely,

Robert D. Pollard

Robert D. Pollard

FREEDOM OF INFORMATION
ACT REQUEST
FOIA-77-2
rec'd 1-7-77

Ed Butcher

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B [illegible]

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

November 17, 1975

[illegible] C. DeYoung, Assistant Director for Light Water Reactors, Group 1, Division of Reactor Licensing

Exhibit 1078

FORKED RIVER PLANT MODIFICATIONS

It has come to our attention that there has been a lack of consistency in the Staff's approach to post-CP safety design changes by Applicants. Since this matter has surfaced at various times and has apparently remained unresolved, the purpose of this memorandum is to provide guidance to the technical staff as to the proper response, from a legal viewpoint, to safety design changes by an Applicant subsequent to the issuance of a construction permit. We believe that it is essential that Staff responses to such design changes take a uniform position with regard to requirements upon both the Applicant and the Staff in reviewing and acting upon such changes from PSAR specifications.

Every construction permit contains language to the effect that the Applicant is authorized to construct a facility "in accordance with the principal architectural and design criteria," which is a reference to the general design criteria of Appendix A to 10 CFR Part 50. It is the Staff's responsibility, in the interim between issuance of a CP and the application for an operating license, to assure adherence to those criteria. Any post-CP deviation by the Applicant from the above authorization of its CP requires a CP amendment, since the Applicant no longer would be in compliance with the terms of the permit. Therefore, we suggest that when notification is received from the Applicant that design changes are planned or in progress, the Staff should request any additional information which it needs to make a judgment as to whether construction is proceeding in accordance with principal architectural and design criteria. If not the Staff should require that the Applicant file an application for a CP amendment.

Should the Applicant refuse to so apply, the Staff should move for an order to show cause pursuant to 10 CFR §2.202, which provides that the Director of Regulation (now the Director, Division of Nuclear

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Reactor Regulation) may institute a proceeding to modify a license or to take such other action as he may deem appropriate by serving on the Applicant an order to show cause. The definition of "license" in 10 CFR §2.4 includes construction permit, so that while the Staff otherwise has no express authority to request that a CP be amended, §2.202 provides a means to achieve the same result, should the Applicant be uncooperative.

We should note that short of a deviation from the principal architectural and engineering criteria, the regulations are devoid of any specific grant of authority to the Staff to require that an Applicant amend its CP to reflect design changes. As the Appeal Board noted in the Cook proceeding (ALAB 129), under normal circumstances, any safety problems arising during construction will await the OL review, and theoretically, if it becomes apparent at the OL stage that there is a safety problem, the Applicant could be required to take corrective action, even if this would entail undoing, at considerable expense, much of what was done during construction. The regulations also seem to reflect the position that the OL stage is the proper time to deal with normal design changes. 10 CFR §50.35 indicates that the issuance of a construction permit "will not constitute Commission approval of the safety of any design feature or specification..." and that no operating license will be issued unless the Commission finds "that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility..."

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To summarize, should the Staff feel that any proposed design change is not within the parameters of the principal architectural and engineering criteria, a letter should be sent to the Applicant by the Director, Division of Nuclear Reactor Regulation requiring the submission of a CP amendment application. Should the Applicant prove unwilling to comply with the Staff's request, the Staff should apply for an order to show cause pursuant to 10 CFR §2.202. However, absent that violation of the specific authorization of its CP, the Applicant may make design changes without applying for a CP amendment at the risk that those changes may cause safety problems which will have to be rectified before an operating license will be issued.

Joseph Gallo

Joseph Gallo
Chief Hearing Counsel

cc: R.Boyd
V.Moore
E.Case
R.Heineman

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