UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Title: BRIEFING ON KEY LICENSING ISSUES ASSOCIATED WITH DEPARTMENT OF ENERGY SPONSORED ADVANCED REACTOR DESIGNS

Location: ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND

Date: TUESDAY, AUGUST 9, 1988

Pages: 1-67

Ann Riley & Associates
Court Reporters
1625 I Street, N.W., Suite 921
Washington, D.C. 20006
(202) 293-3950

DISCLAIMER

This is an unofficial transcript of a meeting of the United States Nuclear Regulatory Commission held on 8-9-88 in the Commission's office at One White Flint North, Rockville, Maryland. The meeting was open to public attendance and observation. This transcript has not been reviewed, corrected or edited, and it may contain inaccuracies.

The transcript is intended solely for general informational purposes. As provided by 10 CFR 9.103, it is not part of the formal or informal record of decision of the matters discussed. Expressions of opinion in this transcript do not necessarily reflect final determination or beliefs. No pleading or other paper may be filed with the Commission in any proceeding as the result of, or addressed to, any statement or argument contained herein, except as the Commission may authorize.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

***

BRIEFING ON KEY LICENSING ISSUES ASSOCIATED WITH

DOE SPONSORED ADVANCED REACTOR DESIGNS

***

PUBLIC MEETING

***

Nuclear Regulatory Commission

One White Flint North

Rockville, Maryland

Tuesday, August 9, 1988

The Commission met in open session, pursuant to notice, at 2:00 p.m., the Honorable LANDO W. ZECH, Chairman of the Commission, presiding.

COMMISSIONERS PRESENT:

LANDO W. ZECH, Chairman of the Commission

THOMAS M. ROBERTS, Member of the Commission

KENNETH CARR, Member of the Commission

KENNETH ROGERS, Member of the Commission

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

S. CHILK

V. STELLO

T. KING

W. PARLER

T. SPEIS

B. MORRIS

SPEAKER FROM THE AUDIENCE:

E. PODOLAK

P R O C E E D I N G S

[2:00 p.m.]

CHAIRMAN ZECH: Good afternoon, ladies and gentlemen.

Today the Commission will be briefed by the NRC's Office of Research on the key licensing issues resulting from the review of the Department of Energy-sponsored advanced reactor designs. The advanced reactors being discussed today are revolutionary designs which are significantly different from the current generation of operating light-water reactors such as the advanced boiling water reactor currently under review.

These advanced concepts are for a 350-megawatt modular high temperature, gas-cooled reactor, a 425-megawatt and 900-megawatt liquid metal reactors. These designs are of the type referred to in the Commission's advanced reactor policy and are included in the proposed rule on standard design certification.

SECY 88-203 provides a detailed discussion of the key issues and provides the background on the staff's proposed criteria to assess the DOE advanced reactor concepts. The key issues are in the area of accident selection, siting source term, calculation and use, adequacy of containment system, and the adequacy of off-site emergency planning.

The paper and today's meeting are responsive to the Commission's decision to provide early guidance to the staff in

the review of advanced reactor concepts.

The ACRS on July 20th of this year provided the Commission their views on the staff's approach to assessing key issues in review of DOE's advanced reactor designs. The Commission is interested in hearing the staff's response to the ACRS letter.

This is an information briefing this afternoon.

Do any of my fellow commissioners have any opening comments they would like to make before we begin?

COMMISSIONER ROBERTS: You say we are interested in what they have to say on the ACRS letter. I would like them to give us a paper answering that letter. I thought there were some thoughtful questions raised.

CHAIRMAN ZECH: I think that is a good suggestion. I had the same comment to bring up during the briefing regarding the ACRS letter, but we can discuss that, perhaps, as we go along.

Any other comments?

[No response.]

CHAIRMAN ZECH: If not, Mr. Stello, you may proceed.

MR. STELLO: Thank you, Mr. Chairman.

The purpose of the briefing this afternoon is to discuss, as we documented in SECY 88-203, a proposal on how to treat the key issues that result from our review of the three DOE reactors. This is an issue, you recall, there has been

considerable discussion on in the past of trying to find a way in which to bring to the Commission significant issues that come out of that review so that the Commission would have the benefit of looking at them.

There is one difficulty in doing this, however, in my view, which we have tried very hard to deal with, and that is to describe issues to the Commission without having the Commission have the benefit of the staff evaluation of each of those reactor concepts. That makes looking at these issues somewhat difficult because you don't see them in the total context in which they should appear when you look at the entire evaluation of the facility and how those key issues relate to it.

We are in the process, and I believe by next month in September we will have at least some drafts of all the SERs, and as I have indicated, when those drafts are ready, we will send those drafts to the Commission, so the Commission ought to at least be aware that when they are considering this matter, we hope to have by September the SERs so that you will have the benefit of looking at the entire evaluation in the context in which these issues appear.

The way in which we have done this is very important. The Commission in its advanced reactor policy statement put out as a matter of policy an approach that I believe was trying to tell the industry don't be constrained by developing a design

which conformed to our regulations, but rather look at how to develop a design which could enhance safety with new ideas and new concepts, and we will worry later about to what extent which regulation ought to apply or ought not to apply; what we are interested in is, in fact, having approaches which have the freedom to come up with new designs which could, in fact, offer significant improvement in safety.

Our approach this afternoon will be to summarize our overall review but to identify as pointedly as we can those very difficult issues that the Commission will have to deal with, and we will be talking about each of those.

On the issue of the ACRS comments, and I think they were very pointed and very thoughtful comments, as Commissioner Roberts has indicated, we intend to describe to you our view of those comments this afternoon. Of course, we have not had the time to prepare a paper or to study them in perhaps the kind of detail we should, but I think we can give you some very thoughtful answers to those comments this afternoon, and we intend to do that as part of the briefing, and we will incorporate in each place as appropriate the ACRS comment that deals with the specific issue.

With that introduction, let me turn to Dr. Speis, and we will begin the briefing.

MR. SPEIS: Thank you, Mr. Stello.

The briefing will be done by Tom King, the Branch

Chief of the Advanced Reactor Branch.

CHAIRMAN ZECH: Thank you very much. Proceed.

MR. KING: I intend to work from the handout. When we put these on the TV camera, we have a lot of information on some of these pages and it didn't come through as clear as I would like, so if you don't mind, we can just work using the written material.

CHAIRMAN ZECH: Will it come on the screen at all?

MR. KING: It will come on the screen. I don't have someone up there now to put it on, but we could put it on if you prefer.

CHAIRMAN ZECH: We can use both. I guess we have passed out copies of the slides?

MR. SPEIS: Yes, sir. Everybody has copies.

CHAIRMAN ZECH: All right. Let's go ahead, then.

MR. KING: I would like to just take a short time up front and go through a little background of how we got to where we are today and what the key features of these designs are, and then get into how we did the review and then our proposals to deal with the key issues.

[Slide.]

As you mentioned, we are looking at three DOE Designs: one HTGR and two sodium-cooled plants. We had provided the Commission back in December of 1986 a paper describing our review plan for these designs. The review will

be wrapped up and documented via an SER. The three SERs are in various stages of preparation at this point in time. We hope to wrap those up in a couple of months.

The purpose of our review of these designs is to provide guidance early in the design process on the licensing requirements for these designs consistent with the guidance in the Commission's Advanced Reactor Policy Statement. This is not a design approval at this stage but rather it provides preliminary guidance on whether or not the designs appear acceptable, whether the designs are heading down the right path or if there is some fundamental problem associated with these designs.

[Slide.]

We had provided you two Commission papers back in the middle of July, SECY 88-202 and 203. Today's briefing is on 88-203, the key issues. Thursday we are going to talk about the standardization issues, paper 88-202.

I might bring up here the first of the ACRS comments. In their letter they mentioned that they felt our paper did not address the full set of concerns, and indeed there are other concerns with these designs. All of the concerns will be addressed in the SER. It was our intent with this paper to address the ones that we felt had policy implications and could affect the viability of the designs.

We agree with ACRS there are other issues. They will

all be described in the SER.

CHAIRMAN ZECH: Since we don't have the screen working this afternoon for your slides, as you go through your papers will you just tell us what page you are on so we can all keep up very easily.

MR. KING: Okay. I am going to start on page 3 now.

CHAIRMAN ZECH: Thank you.

[Slide.]

MR. KING: I did not intend to describe the designs to you today, although I do have some backup information. If you would want to take a few minutes and go through the designs, we could do that. What I did want to mention were the key features of the designs. These designs were described to the Commission back in October of 1986.

With regard to standardization, all three designs are what you call modular concepts. They involve reactor modules, multi-modules on a site to produce power, and there would be various options as to whether you could put four modules or eight modules, whatever, depending on the power needs of the utility. The goal of all three designs is to have them certified. They would be standard. And they concentrate all their safety functions in the nuclear island. Therefore, they have tried to make a distinction about the balance of plant nonsafety-grade.

[Slide.]

On page 4 regarding the safety features of the design, the key features are their passive decay heat removal and passive reactor shutdown capability. This provides them with the potential to prevent core damage under severe challenges such as ATWS and station blackout and loss of heat sink events and so forth.

I should mention here that even though these are paper reactors, there is substantial amount of R&D and experimental basis for the claims that are being made.

For the liquid metal reactors, ERR-2 has operated for a number of years with metal fuel and has demonstrated some of these inherent shutdown characteristics. More demonstration is planned. For the gas-cooled reactor, the fuel retention capability has been demonstrated with success at the Fort St. Vrain and the success of the German HTGR program, which uses essentially similar fuel. The German program has also demonstrated the capability for passive reactor shutdown and decay heat removal through a steel reactor vessel.

So I wanted to make it clear that although these are paper designs, we are not just talking about wishful thinking here; we are talking about some things that have some basis in fact.

Also key to these designs is their intent to reduce the need for operator action and reduce the potential for the operator error to effect performance of safety functions.

Because of these safety features, the designs proposing that they be sited using a mechanistic siting source term, that they do not have a conventional containment building and that the need for off-site evacuation planning can be reduced.

Page 5 --

CHAIRMAN ZECH: Before you go off page 4, the ACRS letter to the Commission noted that there is very little said about operation and staffing, of the advanced reactors' requirement for operations and staffing.

Do you intend to do any more in developing requirements for operator and staff regarding advanced reactor designs? In other words, where does that stand? There wasn't much focus. It seemed to be a reasonable focus on designs, at least conceptual designs, but there wasn't much focus at all on operation focus.

I think we have learned in our experience in the past with light water reactors that it is important to focus on the operational side of it. We know that human errors are there. We focused on human factors recently. We recognized that the operators make mistakes even if they are well trained and what it our focus on operational and staffing? I think the ACRS question in this regard was a good one.

Could you comment on that?

MR. KING: Let me answer it in two parts. We looked at operations from the standpoint of human error. We looked at

the design as to what does the human have to do and what could the human do to negate the safety features of the design. We did that at this stage of the review and we were very impressed by the steps taken by the designers to eliminate the potential for operator errors to affect the safety of the plant as well as to reduce the number of actions the operator has to take.

CHAIRMAN ZECH: Do you know to what degree the designers involved operators in the design?

MR. KING: I do not know the extent to which they involved people with operating experience in their design teams.

CHAIRMAN ZECH: It might be good during your further reviews in the future to ask them about that. I would suggest that you recommend that they have them involved in it. Sometimes they can make contributions. I appreciate the fact that designers have learned things about operations in the past too, but I think involving the operators directly could be beneficial.

MR. KING: I'll do that.

The second half of the question that ACRS raised had to do with the number of people required to run these plants. What has been proposed by DOE is essentially one operator for two or three or four modules.

We felt that that issue was not key to the viability of these designs. It was pretty much of our planning to put

that off until a later review stage, although we do agree it is an issue. ACRS felt it was an issue because we do have requirements on the books regarding the number of operators required to be in the control room and so forth.

We felt at the conceptual design stage that was not an issue, the number of people was not an issue key to the viability's design so we did not spend a lot of time on that particular part of the ACRS concern.

MR. STELLO: Let me shorten the answer. The thrust I think of the ACRS is they are raising a question as to how many people do you really need to operate the plant. The design was developed on the basis that very few people were needed to run the plant, relative to today's light water reactors. We think that they have made the case, the passive systems I guess -- what are they, 30 hours without any operator intervention?

MR. KING: 36 -- at least 36.

MR. STELLO: 36 hours in the event of an accident and you don't need any operator action -- you know, those kind of criteria were used, which at least suggest that the need for a large staff to deal with incidents and accidents has been by design designed out of the plant.

But our view is very simple, that if in fact you need additional staffing and it is demonstrated we need them, that clearly is something that could be added at any point in the process, but we applaud the concept for building into the

design something that makes it very easy to operate without need for a lot of operators to run the plant.

We don't think there is any problem in raising that issue later if someone really is going to develop this design and market it and someone buys it. that is something we feel very comfortable with as an issue that can be put off, but to recognize at the beginning they have done a very good job of trying to minimize the need for that staffing as well as for maintenance.

CHAIRMAN ZECH: I think that seems reasonable but I can't help but comment it has been my experience in my earlier career that designers sometimes have a tendency to not recognize the number of people will be needed to operate their equipment. They frequently, at least it has been my experience, tell those who are interested in their product that very few people can operate it. It is not necessary to have any more than a very few and then experience shows that's not true.

All I say is it may not be important enough to get into the details of that aspect of it at this time but I would just caution the staff to please take a realistic look at the number of people that you might need in any kind of an emergency situation or any kind of a normal operating situation.

It has been my experience and I could go into more

detail on it but I won't this afternoon that very often you design something and you do not factor in a realistic number of people that's needed. Now since we have had operational experience in this agency and we recognize the importance of people and I think it behooves us to at least emphasize to the designers this cautionary comment from time to time because there is no sense in not learning from past experience and I do think that it is worth careful thought, even early on.

MR. KING: I will mention that each design team did have utility involvement in reviewing the design. I suspect that brought in some operational overview as to workability.

CHAIRMAN ZECH: Should have been helpful. Good. I hope there were utility operational people.

MR. KING: That I can't answer at this point, but your point is well taken.

CHAIRMAN ZECH: All right. Thank you. Let's proceed.

MR. KING: Page five.

[Slide.]

MR. KING: Let me briefly summarize how we approached the review, following with the guidance in the Commission's advanced reactor policy statement which said that the Commission expects as a minimum at least the same degree of protection of the public and environment that is required for current generation LWR's. By current generation LWR's we mean

those standard plants that are currently under review in NRR today, the ABWR, the SP-90 Westinghouse design and the System 80 Plus CE design.

The policy statement went on to say that furthermore the Commission expects advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent passive or other innovate means to accomplish their safety functions.

Now starting with that as our guidance, we had to look at the process that we want to use to show that we have at least the same degree of protection in this current generation of LWR's. The way we approached that was to first look at what's on the books for LWR's, the regulations, the standard review plan, the various policy statements; make an assessment as to which of those are applicable to these advanced designs; then, for the things that are not applicable -- for the unique attributes of these designs -- come up with some recommended criteria that we felt would provide equivalent protection.

[Slide.]

MR. KING: At page 6, we also in addition to that looked at how we would want to address the expected enhanced safety portion of the Commission's policy statement and both of these -- the next several slides will talk about both of these.

Let me move on to page seven.

MR. STELLO: I think it -- this is an issue the ACRS raised also and I think in their letter, if anyone has it, on

page 5 of their letter they raise the question of how safe should the plants be.

I guess I read their comment of suggesting that perhaps the staff was trying to rachet beyond what the Commission's intent was in their advanced reactor policy statement. I don't think we've done that. These reactors are in fact being designed without certain kinds of safety features that current reactors have. They therefore have built in certain additional features in my view that form the basis upon which you can possibly come to the conclusion that we could take a different approach and I think that is something the Commission wanted designers to consider, but the suggestion that the staff is ratcheting beyond what the Commission proposed is a comment they make and I just don't think we have done that and I don't know how you resolve that. That seems to be the impression they have. These designs are designs as proposed by DOE and it is complicated because of the trade-off they have made.

I think we consciously have tried to implement the Commission's guidance as best we know how. I don't think we have done that.

CHAIRMAN ZECH: Okay, fine. Thank you. Let's proceed.

[Slide.]

MR. KING: On page 7, regarding looking for the same

degree of protection, I wanted to emphasize that we did not rely on a single measure of safety in looking at these plants, such as comparison of PRA's to LWR's or comparison to safety goals.

What we did was we evaluated the various factors that contribute to safety on an LWR and require that these plants have the same factors or develop factors that we judge would provide the same degree of protection or defense in depth.

Pages 9 to 15 of this handout provide a summary of those factors that we looked at in the comparison between LWR's and the DOE advanced reactors. I am not going to go through every one of those pages but I will use page 9 as an example, and these pages also show where the four key issues that we are going talk about came from in doing this comparison.

[Slide.]

The other key feature of our review was prototype testing. We felt that all three of these plants were different enough and we're proposing major changes to the way safety functions are accomplished and prototype testing is essential to verify that they are capable of withstanding the challenges and capable of limiting the release of fission products to what designers say they can --

COMMISSIONER ROBERTS: Does DOE have the same point of view?

MR. KING: Two of the three designers have the same

point of view. The MHTGR, although they are planning to build a demonstration plant, they do not feel they need any special safety tests to demonstrate fission product retention capability. We differ with them on that part, but the other two, the sodium plants have the same point of view.

COMMISSIONER CARR: As I read it, you were only going to prototype those things that were unique in these --

MR. KING: No.

COMMISSIONER CARR: The way I read your paper --

MR. KING: No, the intent was to prototype at least one module.

COMMISSIONER CARR: Complete. Full-scale.

MR. KING: Full-scale prototype -- may not have had a turbine on it but it would at least go out through the steam generator and run it through a series of tests that would demonstrate these passive safety systems work the way they are intended to work and any other unique feature on the plant.

COMMISSIONER ROBERTS: Would all of that be done by DOE?

MR. KING: It could be done by DOE. It could be done by a combination of private sector, Government initiative. It could be done by a private initiative. I think that is not settled at this point.

MR. STELLO: I think I ought to add at this point that I guess a week or two ago DOE announced that they were

going to build two new production reactors. One of them in Idaho was to be based on HTGR concept, I think with modifications.

They propose to modify it, in fact I think are proposing to add a containment kind of a structure. I am not sure I know in detail what that means and intend to build those out at Idaho. To what extent they would include as part of that proposal prototype testing I don't know. It could easily be part of the program that they are going to develop for the production reactors.

COMMISSIONER ROBERTS: Have they announced and designated the two production reactors and where they will be?

MR. STELLO: As I understand, yes. My recollection is that one, the HTGR, is in Idaho and the heavy water is in Savannah River.

COMMISSIONER CARR: But it is not unreasonable that if they build the HTGR out there it might serve as a prototype.

MR. STELLO: That was precisely my point, that somehow that question will now get answered for a different reason, but I'll come to grips with it because they have announced those two selections.

MR. SPEIS: I think it should be mentioned that based on our reading, there are some substantial differences, you know. In addition to the containment, it uses enriched fuel. The passive systems are different than the ones we are

reviewing but we haven't reviewed them, of course.

MR. KING: I should mention also that one of the ACRS comments was that our proposal was only for modest prototype testing. I don't think that is true. I think our proposal was for fairly extensive prototype testing. I want to make that clear. When we get to pages 16 through 19, we put in some more definition of what we had in mind for the prototype tests, so maybe when get to those pages, we can talk about that a little bit more.

CHAIRMAN ZECH: Before you go off, where are you now, on page 9?

MR. KING: I am ready to go to page 9, yes.

CHAIRMAN ZECH: You can ahead. Then I have a question for you when you get there.

[Slide.]

MR. KING: Pages 9 through 15 summarize the key factors that the staff looked at in assessing the DOE concepts in comparison to light-water reactors. The purpose of these pages is to illustrate how the review was conducted, how licensing guidance was developed, how the key licensing issues were identified.

Let me just on page 9, as an example, talk about the item we have labeled Accident Prevention. LWRs through the regulations are required to be designed to certain conservative codes. For example, reactor vessels to ASME Section III, Class

1. We looked at the DOE plants to see if they were designed to more conservative codes and put down as a requirement that that was a requirement they had to meet where they had high temperature conditions, for example, that a light-water reactor did not have.

We are requiring that they provide a high temperature code case to cover their conditions. For service levels, when the reactor vessel is designed and has various service levels of stress limits, temperature limits that it can go to depending on the severity of the event that is being analyzed, we are looking for equivalency on the DOE concepts versus the LWRs. For example, on one of the designs we felt it went to the higher service levels on too frequent an event. That was one of the design issues we raised on that plant, to try and reduce the frequency of those severe challenges to the reactor vessel.

Quality design, construction, operation and maintenance requirements that would apply to LWRs apply to the DOE concepts. On page 9 here it says "with possible enhancement in certain areas." For example, the HTGR. One of the key features to that plant is the fission product retention capability of the fuel, the fuel quality. It can maintain its integrity under high temperatures, and the designers are proposing that the quality of that fuel coming off the production line is so good that there is very little fission

product release even with severe challenges to the plant. We feel maybe in that area there might be some enhancement needed in terms of quality assurance to ensure that that fuel does come off the fabrication line and is examined on a sample basis to ensure its quality. So that is an example of what I had in mind with these words on page 9.

CHAIRMAN ZECH: That was the question I had. You just answered it. But you do intend in that example you just gave to have more rigorous quality assurance requirements, perhaps, if it looks like that is appropriate. That is what you are saying.

MR. KING: Yes.

CHAIRMAN ZECH: I think it is important that we be mindful of that. In these advanced designs, quality assurance may be necessary to enhance them if they are counting on some new or improved type of process.

MR. KING: That's correct.

CHAIRMAN ZECH: Thank you. Proceed.

MR. KING: Go to page 10.

[Slide.]

I just want to point out in going through the next several pages where the four key issues came from. On page 10 we talk about the factors that contribute to plant protection and mitigation. We are essentially requiring equivalent reactor shutdown capability and decay heat removal capability

as LWRs. When it came to the containment building, our proposal is to substitute a high degree of core damage prevention to be verified by the prototype test for containment building. We will get to the specific proposal.

[Slide.]

Jumping to page 12, another of the key issues came in looking at the safety analysis aspect of the design. For light-water reactors they are designed for a range of anticipated operational occurrences of design basis events, plus with the severe accident policy statement, the staff is considering how to look at severe accidents for future LWRs. For the DOE concepts, we also have required essentially equivalent treatment in terms of identifying and analyzing anticipated operational occurrences and design basis accidents, with a range of more severe challenges being added to compensate for the designers wanting to use a mechanistic siting source term calculation.

In other words, in light-water reactors, siting source term is assumed that it's derived from an assumed core melt event. Since these plants are trying to prevent core melts, prevent severe core damage, to compensate for that we felt they needed to look at a range of events beyond the traditional design basis envelope when they were looking at siting for these plants, and we will come to the specific proposals in these areas as we get through the presentation.

[Slide.]

On page 13, siting I just talked about. Emergency planning. These designs, all three have long response times. What we felt was perhaps we should give some credit to these long response times, and perhaps the need for the detailed pre-planned off-site evacuation could be compensated for by having a lot of time prior to the need to evacuate people. So we have proposed some criteria that we feel provide equivalent protection in that area.

CHAIRMAN ZECH: Are you saying that you currently plan to have emergency plans developed but not require any evacuation exercises? Is that how I understand the thinking?

MR. KING: That's right. They would not require early evacuation drills or early sirens.

CHAIRMAN ZECH: But you have emergency plan --

MR. KING: It would be an emergency plan.

CHAIRMAN ZECH: -- in place.

MR. KING: That's correct.

CHAIRMAN ZECH: In case? Is that right? If necessary?

MR. KING: Yes.

CHAIRMAN ZECH: Why do you figure you wouldn't need any exercises?

MR. KING: We feel the time that is provided in these designs before you would get any significant release of

radiation would give local authorities sufficient time to move people without having a pre-planned evacuation and sirens.

CHAIRMAN ZECH: I see. In a routine-type situation.

MR. KING: In a routine like response for chemical skills or things that happen every day in the country.

MR. PARLER: That suggests to me, Mr. Chairman, that there would not be any off-site emergency planning requirements to speak of.

CHAIRMAN ZECH: That is what I'm trying to find out. I think, as I understand it, there would be emergency planning requirements, is that right; but you wouldn't require exercises? That is what I understood. Maybe I'm not correct. Is that correct?

MR. PODOLAK: Mr. Parler is correct.

CHAIRMAN ZECH: Who is answering back there? Please step up to the microphone and identify yourself for the reporter.

MR. PODOLAK: Ed Podolak from NRR. I contributed to the Special Paper.

The concept is that there would not be off-site emergency planning in the traditional sense that we know. There would be plans that would include off-site responders who would respond on site, like fire and police and things like that, but there would not be plans for off-site response in the traditional sense beyond notifications. In other words, the

off-site authorities would be notified and there would be procedures in the on-site plan for doing that. They would be afforded the opportunity for training if they wished, but it would not be a traditional emergency plan as we know it.

I hope that is clear.

CHAIRMAN ZECH: Let me just see. You are saying that you do not envision off-site emergency plans.

MR. PODOLAK: That's correct. That is what the paper --

CHAIRMAN ZECH: With these new advanced designs.

MR. PODOLAK: That is correct. In other words --

MR. STELLO: I think maybe what we ought to say is in the context of our current Appendix B requirements, which are very specific elements of emergency planning that require early notification and other elements that have to be implemented. For example, the idea of the ingestion pathway as an issue which is now in our regulations, we don't see that that would change. You have that element. But because of the inherent features and the very long times -- For example, 36 hours leaving a reactor untended in the event of an accident allows a lot of time, to where this early response that is inherent in our Appendix E now could be done in a way different.

So in terms of meeting the current Appendix E answers, we would advocate that -- If everything they have said is true and the design, in fact, does what it is advertised to

do, you would not need to have that kind of response. That is one of the key issues to the Commission and one of the issues that the ACRS commented specifically on.

CHAIRMAN ZECH: Of course, that is a very fundamental decision that will have to be made, and I am sure you are going to be looking at it very carefully.

MR. STELLO: That is one of the key policy issues that comes out of this.

CHAIRMAN ZECH: All right. Let's proceed.

MR. KING: Let's move on to page 16 and talk about the prototype reactor test.

[Slide.]

At this stage of the review what we were trying to get in place is the concept of the prototype reactor test. We are not trying to at this stage pin down exactly what the prototype will look like or exactly the tests that will be performed, but to give enough guidance regarding the types of tests and the scope of the plant that we feel is important to test that we can proceed into the next stage.

Again I mention ACRS had commented that we only had modest requirements for prototype testing. We feel that what we have is more than modest requirements. Let me just go through it quickly.

On page 16 what we had in mind was at least one module, full size, would be tested with its associated

instrumentation and controls and other supporting systems, at least through the steam generator. It would have to be built to the same standards and design as the plant to be certified. If it did not include the whole plant -- for example, the turbine generator -- they would have to be able to simulate any interface requirements that could be important to testing, important to the safety of the plant. They may require some special instrumentation or features to be on the testing.

COMMISSIONER CARR: But that doesn't preclude somebody from building a reactor as a prototype and then turn it into a commercial production plant.

MR. KING: Absolutely not.

COMMISSIONER CARR: Okay.

[Slide.]

MR. KING: As far as the test program goes, the purpose would be to verify the plant response to these bounding and challenging events and generate sufficient data so that the analytical tools used on the design can be verified. It would be the kind of test that would be conducted in a stepwise fashion, from low power, low decay heat on up to higher power and higher decay heat conditions so you would get some confidence along the way before you proceed to the next level of severity that things are going the way they are supposed to go.

The test program would be directed toward looking at

internal events. We have no intent to test things like earthquake, tornado and so forth. We don't believe that testing that damages the plant is necessary. We feel that through this stepwise process and through running tests, that you don't have to damage the plant, and that would allow a utility, an organization to then turn the plant over to an operating utility to use it for power producing purposes at a later date.

At the conceptual design change, we essentially have told DOE that at the next review stage they propose the detailed test program for our review and approval. We wouldn't certify the design until the prototype test results were available and reviewed and accepted.

[Slide.]

[Slide.]

Pages 18 and 19 are an example test matrix of the kind of things you could run. They would be directed toward the more severe events like ATWS events. You would look for negative temperature coefficient, various modes or loss of decay heat removal type events, whether it is loss of heat sink or station blackout and so forth. For those events that are dependent upon burnup conditions, you would run those at several times up to the equilibrium burnup cycle, equal to the burnup level of the core.

Then the prototype would also verify much of the

other plant conditions, whether the insulation design is proper, the shielding design is proper, things like that. There would be a lot of other benefits that come out of prototype testing, including the operation of maintenance benefits.

[Slide.]

Page 20, the approach to addressing expected enhanced safety requiring each of the designers to document where they feel they have enhanced safety characteristics or margins over and above what the minimum or the adequate protection level is.

We are looking at the designs to see if they should implement any of these additional enhancements, either if the margins to what's adequate are small or if some changes could be justified on a cost-benefit basis.

ACRS had made a comment on the cost-benefit, that they didn't feel this was a workable approach. The staff had some experience in the past, particular the GESSAR-2 review in looking at design changes on a cost-benefit basis.

If we find that they will be unworkable, we could drop it at a later date but at this point, we still feel that it is a workable approach.

[Slide.]

All right. On page 21, let's talk about -- let's start with the first of four key issues. That deals with accident selection. What range of events should these plants be required to be analyzed for -- to look at. We have

essentially defined four event categories. We've tried to stay away from the traditional terms, design basis accident, beyond design basis accident and so forth because we found it caused a lot of confusion.

So, we defined four event categories, the first one being those events that are normally expected during the life of the plant equivalent to operational -- anticipated operational occurrences. They would be analyzed just as they are for LWRs and those events in terms of release criteria. The plants would have to meet the 40 CFR 190 and 10 CFR 50 Appendix I Release Criteria, essentially the ALARA type criteria.

[Slide.]

MR. KING: Event category II is something that's equivalent to what we on LWRs -- we traditionally call design basis accidents. Page 22. They'd be selected using engineering judgement with the benefit of insights from a PRA. We traditionally look at events down to about ten to the minus four per year if you're looking at things in probabilistic terms and would include a lot of the traditional events that are on light water reactors, site suitability, say shutdown earthquake and loss of power, that kind of thing.

These events would be used in a citing determination as we'll get to in a couple of pages from now.

COMMISSIONER ZECH: How is the Commission's safety

goal policy used in your considerations of the severe accidents for advanced reactors?

MR. KING: Event category III is where the safety goal policy was used as the basis for setting the limits on event Category III. If you turn to page 23 --

[Slide.]

COMMISSIONER ZECH: Could you talk about that just a little bit? I see you've got that in there but could you elaborate just a little bit on how you did that?

MR. KING: Okay. What we did was we took the performance guideline. The ten to the minus six of a large -- probability of a large release performance guideline from the safety goal. We said, event Category III, the severe accident event category, should ensure that that performance guideline is met. To do that, we said we ought to look at events down to approximately ten to the minus seven per year if we're thinking in probability terms to ensure that the cumulative effect of those would not exceed the ten to the minus six performance guideline.

So when you're defining event Category III, we in a probabilistic definition put a ten to the minus seven limit on it. That shows up at the top of page 23. In addition to that, because when you get down into those kinds of low probability ranges, there is a lot of uncertainty, a lot of judgment involved, we also put on what we call the set of bounding

events that would be based upon engineering judgment that would tend to bound the uncertainties of some of these low probability events that we felt with this combination of events selected from a PRA that go down to the ten to the minus seven and it's good engineering judgment.

We felt we could show that that ten to the minus six safety goal performance guideline would be met and that's basically the definition of event Category III.

COMMISSIONER ZECH: All right.

MR. STELLO: Here's one of the key issues and you can see how it relates directly. In light water reactors, you have a Commission safety goal policy saying ten to the minus six should be a goal with respect to a significant release. In order to get a significant release, you by definition have to have some failure of containment and remember, these plants are being designed without containment.

So, what this basically is getting us to is a criterion that say you won't have any core damage at ten to the minus six which is a basis then to say in order to get the balance of saying, well, if you're going to make the argument of not providing containment systems, you are going to have to meet a criterion which says basically --

COMMISSIONER CARR: Conventional containment systems.

MR. STELLO: Conventional containment systems. You are not then -- by definition are going to have to move out to

where you can't damage the core. That's how we get there.

COMMISSIONER ZECH: All right. Let's proceed.

[Slide.]

MR. KING: One of the ACRS issues was that the staff should require more safety than is called for in the safety goal. We feel that by defining event Category III the way we have, we are being consistent with the safety goal.

COMMISSIONER ROBERTS: You don't think you're more demanding than the safety goal?

MR. KING: No. I don't think we are. I think we may be conservative in showing that it's met but the intent is not to be more demanding than the safety goal.

COMMISSIONER ROBERTS: Well, let me tell you how I think the debate can happen and I see where the ACRS could say in conventional plants, core melt frequency is ten to the minus four. At this plant, core melt frequency is equivalent to ten to the minus six. That's two orders of magnitude more restrictive than currently in light water reactors and that's a fair comment but when you use as a possible issue ten to the minus six and no significant release in order to get there, you by definition have to move the core melt frequency up to ten to the minus six because you don't have the conventional containment system. I could see the argument.

COMMISSIONER ZECH: All right. Let's proceed.

MR. KING: Another one of the ACRS comments was that

we should -- judgment is needed in looking at events, phenomena or sequences with large uncertainties. You have to have permission for engineering judgment to accommodate those. We feel that our definition of Category III provides such provision.

Finally, on this item going back to page 22, ACRS had made a comment on Category II where we said that in looking at those events, we use conservative analysis and we use the traditional conservatisms that are applied in looking at those similar events for LWR such as single failure criteria, no credit for non-safety grade equipment and ACRS made a comment that we should -- should permit some credit for non-safety grade equipment if it's justified. We agree with that. We didn't mean this to be such a rigid requirement that if there was justification we wouldn't allow it. We would allow some credit for non-safety grade equipment if justified.

COMMISSIONER ROGERS: Excuse me. The use of that word conservative analysis is something that I brought up a couple of weeks ago or so at a meeting here where we were talking. I see that the same interpretation was made by or recommendation made by one of your review committees pointing out that the way to approach it is to do a realistic evaluation -- a realistic analysis plus a specified margin on top of that rather than to do an analysis which involves a worse case situation at every stage and call that conservative.

The distinction between those two approaches is, I think, an important one and I'm just wondering what you have in mind when you say conservative here. What your approach is.

MR. KING: What we have in mind essentially is doing analysis out to two standard deviations, for example, and analyzing core. You'd apply hot channel factors to the core that would account for flow maldistribution, types of effects that could cause fuel cladding temperatures to be hotter than you would. In analyzing fuel pin cladding temperatures, you would put those conservatisms into the calculation and use that in determining whether you meet the acceptance criteria or not. Whereas when you move into event Category III where we allow best estimate analysis, you would do similar analysis without the hot channel factors and see how much margin above the acceptance criteria you have.

What we're trying to do, we're trying to keep event Category II both the events that are in there as well as how they are treated equivalent to what's done on LWRs so that we can show that one for one equivalency as much as possible and when we move to event Category III which go beyond the traditional design basis envelope, we switch to the best estimate analysis, which is done traditionally on LWRs and looking at PRAs and severe accident analysis that go beyond the design basis envelope.

So, we're trying to keep equivalency in terms of

events and the way they're treated as much as possible.

COMMISSIONER ROGERS: Okay. I don't want to take us too far afield on this particular point right now, but I think it'd like to learn a little more. Thank you.

COMMISSIONER ZECH: All right. Proceed.

MR. KING: The last event category at the bottom of page 23 is a set of events that would be used in assessing the need for emergency planning. Again, the intent there would be to look at a similar range of events as was looked at for LWRs and those were defined in NUREG 0396 LWRs.

[Slide.]

MR. KING: Okay, moving to page 24, how do we use these event categories. The first place they show up is in looking at what do we use for citing source term for these plants. As I mentioned earlier, these plants have the potential to prevent core damage instead of imposing a deterministic, large release on these plants, we wanted to give them credit for these inherent safety characteristics. To do that, we felt if we could define a range of events that they had to look at for citing purposes, that we wanted to define it so it would provide equivalent protection is what's used today on LWRs.

So, what we're proposing is for citing purposes, that they look at events in event Category II in a mechanistic analysis and have to show that they meet 10 percent of the Part

100 Dose Guidelines and look at the events defined in the EC-III event category and show that they meet 100 percent of the Part 100 Guidelines.

We came up with a 10 percent number. If you look at the way light water reactors analysis their design basis accidents, the accidents that are analyzed in chapter 15 of the FSAR, wherever they are allowed to use a mechanistic analysis, the standard review plan that exists today requires that they -- that the releases from those events be limited to what they call a small fraction of Part 100 which traditionally is either 10 or 25 percent of Part 100, so we chose the 10 percent value and imposed that on the EC-II events for the DOE plants.

The other thing we would require the designers to do is to look at their EC-II and EC-III events and make sure that they're not sitting on some cliff where a small change in the event could all of a sudden cause a large release. There would have to be some adequate margin beyond -- the large releases beyond the events that are actually being analyzed in EC-I, EC-II or III.

Again, for the meteorology, it would be assumed in calculating compliance with the Part 100 Guidelines, conservative the same as on LWRs.

[Slide.]

MR. KING: Containment question, moving to page 25. The proposals from DOE essentially say the containment building

is not the only way to retain fission products. They can be retained in the fuel. They can be retained by preventing accidents. What we tried to do is come up with a set of criteria that would define those things that would have to be met for us to accept the design without a conventional containment building.

There are six of these criteria, the first one being that they'd have to provide multiple barriers to radiation released, by that we mean more than one that would prevent -- which would cause the plant to meet the release guidelines for event categories one through three.

The second item is, we would have to demonstrate acceptable plant performance over a range of sever challenges, using a remotely sited or isolated sited full-sized prototype reactor. That they would have to provide additional enhanced QA surveillance, inservice inspection and so forth as necessary, to ensure that the system's structure and components which contribute to performing the containment function, are in fact capable of performing their function over the life of the plant. The MHTGR field is the good example to illustrate that point.

The design would still have to provide protection of safety related systems, structures and components from sabotage and external events equivalent to that for LWR's. ACRS made a comment regarding the sabotage issue. They felt there was a

need to develop some guidance for advanced reactor designers in the area of sabotage. I think that's a point that is worth pursuing further with ACRS. We would intend to do that at a later review stage.

COMMISSIONER CARR: That remotely-sited requirement bothers me a little bit. If you built this inside a conventional containment, that would take care of that, I would assume.

MR. KING: We looked at that option, yes. We didn't put that up as the favorite option because then you're really not testing the plant that you want to certify.

COMMISSIONER CARR: You put the same plant inside a containment.

MR. KING: As long as you're not impacting the passive decay heat removal system

COMMISSIONER CARR: All I'm saying is that remotely-sited means different things to different people.

MR. STELLO: If you put the containment on and suppose that has the certified design; that would obviously be no problem. If, however, you put the containment on and you may not be able to test then, the actual passive features; then you have a problem.

COMMISSIONER CARR: Why?

MR. STELLO: Because the --

COMMISSIONER CARR: Why wouldn't you be able to test

the features? The reactor and the containment ought to be separate.

MR. STELLO: What we explained, the reliance on being able to remove by natural circulation and availability in the --

CHAIRMAN ZECH: Why don't you explain a little bit more the design.

COMMISSIONER CARR: I understand that, but I mean, to just say that the guy has to go build it out somewhere away from everybody may be awfully hard to do. There may be ways to get around that.

MR. STELLO: Okay.

MR. KING: As long as the design that you were testing verified the one you wanted to certify, I would have no problem with that.

COMMISSIONER CARR: Okay.

[Slide.]

MR. KING: Moving on to page 26, the fifth item would require the designers to eliminate core melt significant positive reactivity feedback or other accidents with the potential of a large radiation release from the EC I, II and II event categories. The point there, we were concerned about was once the plant gets into a condition where you begin to worry about the integrity of the core or a core melt situation, we felt there was enough uncertainty in analyzing and

understanding those events that we didn't want to get into a situation of making an argument that you had in-vessel retention and you didn't have re-criticality concerns. We felt to accept the design without a containment building; we just waned to eliminate those from consideration.

ACRS raised comments in this area. One of their comments was that staff criteria did not require diverse and passive safety systems for the reactor shutdown and decay heat removal. We did require diverse systems for shutdown and decay heat removal, but we did not put the requirement in our proposed criteria that the systems had to be passive.

We felt that that might hamper design innovation, maybe a little too much. That there might be other ways that they could get decay heat removal and reactor shutdown systems. We did not put in a specific requirement that they had to be passive.

The last item on page 26 had to do with the looking whether a conventional containment building could be added and be justified on essentially a cost benefit basis. We took a look at the designs at the conceptual design stage from that standpoint and did not find that we could justify them on a cost benefit basis.

ACRS commented in general on the whole containment issue, that they are not ready to accept arguments of accident prevention and protection as justification to eliminate the

containment building. They went on further to state that they could decide otherwise in the future in response to a persuasive argument. We feel that prototype testing is the persuasive argument. With that requirement on these designs, we'll have confidence that they can preform the safety functions and retain the fission products the way they're being advertised to do today.

I'm not saying that we disagree with ACRS, but we do believe the persuasive argument will be the prototype testing.

COMMISSIONER ROGERS: Could I just make a comment on that. What is your proposal for how long a prototype test should take? We're still learning things about LWR's which have been in service for 30 years and it seems to me that argument is one that you've made that has to be looked at a little bit, because how much experience will you get with a prototype test? You'll certainly get very important experience, but we know that there's always something else out there that we haven't quite looked at or experienced that comes up.

When you are making the argument that the prototype test is a total substitute for a containment, I think you have to really convince everybody that that prototype test has really just about exhausted all the possibility.

MR. KING: You're right. There has to be some judgment and what we propose at the conceptual stage is taking

the prototype up to the point where you reach equilibrium core burn up, because the reactivity coefficients are burn up dependent. The decay heat distribution in the core is burn up dependent. those are key features to the key items that you have to look at in showing that these passive safety features work.

For these plants, the DOE plants, that would probably be in the range of 2-3 years that you'd have to operate the prototype. We do not intend the prototype to go on and demonstrate component reliability or other things. It's to demonstrate the performance of those key safety features of the plant, up to and including the equilibrium conditions that they would reach as the plant proceeds forward.

COMMISSIONER CARR: You're talking one core life probably.

MR. KING: Yes, which for these plants is probably 2-3 years.

COMMISSIONER CARR: That's probably before anybody would start construction on something anyway, until the design was certified.

MR. KING: Yes.

COMMISSIONER CARR: Then, while they were building it, you would still be operating the prototype for another five years while they're building it, would be my guess.

MR. KING: If it's turned over to a utility, sure, it

would keep going until --

COMMISSIONER ROGERS: Well, that might be an --

COMMISSIONER CARR: Even to a design.

COMMISSIONER ROGERS: -- important point to consider, though, in this whole thing and that is that it sort of touches on the question that you raised about, could you turn the plant into a power plant, once it's passed its prototype state. It might be that that period after you've decided that -- or everybody's decided that it looks good enough to go ahead and use it as a basis for a license to build a plant, maybe it still ought to run in that construction period to start to look at any of these other things.

That might be something to be considered or the value of keeping that thing going up to the point that it's Mark I realization is actually ready to run as built on the basis of the prototype. The other point that I just wanted to raise a question on was, in the bullet just above that on page 26, the significant positive reactivity feedback question. Isn't it true that in some of these designs, you get a negative reactivity feedback as an integral measure of performance, but that's it's the sum of a number of different things that are possible and happening in there, that you don't measure individual. It's just the total effect.

I wonder what your thinking is about those one-level-down contributors to the net reactivity feedback which may be

negative net, but has some positive contributions to it -- liquid metal reactors with a positive void coefficient. For example, if you add it all up, it's a negative, but there is a big positive contribution to that net negative. I always worry about something that represents a difference between two big numbers that turns out to be either negative or positive, a little bit.

I don't know if that's the case here, but it seems to me that one has to look at the individual contributors to that.

MR. KING: The intent of these words was not to limit some individual contributor from being a positive contributor. The intent was that when you look at the net, it is negative. For the liquid metal reactors, sodium void net is positive, by the way. That's the kind of event that we want to eliminate from consideration in these designs to eliminate it and reduce its probability. The prototype testing would be used to demonstrate that certainly the net effect of these feedbacks is negative and as much as possible, separate out the various components.

COMMISSIONER ROGERS: Well, if you could isolate them, so that one doesn't suddenly become a dominant feature.

CHAIRMAN ZECH: All right, let's proceed.

[Slide.]

MR. KING: Page 27, the last of the key issues, the offsite emergency planning -- our proposal there is that

traditional offsite emergency planning would not have to include early notification, detailed evacuation plans or provisions, detailed evacuation provisions for exercising the plan provided two conditions are met.

The first one is that for the event categories I, II and III which have to be considered in the design, all events in that category, you'd have to show that you did not exceed the lower level protective action guidelines during the first 36 hours following the event.

The second one deals with more residual risk concerns. It says that when you look beyond event categories I, II and III, that you don't have a lot of things out there that could represent a significant contributor to risk that would cause you to exceed in the first 36 hours, the protective action guidelines. We've essentially said that in looking at those events, that the cumulative probability of those has to be less than 10 to the minus six per year.

ACRS indicated in their letter, general agreement with these two criteria. I might mention just quickly where the 36 hours came from. Upon looking at historical evacuations and how long it took where there was no pre-planning, most of those were able to take place in 2-8 hours. We felt that if we'd allow 24 hours, a full day, which includes daylight and nighttime, that ought to be sufficient time to conduct an evacuation without having pre-planning to the extent that

exists on light water reactors today.

On top of that, we added 12 hours for the plant operating staff to diagnose the event, understand what's going on and initiate the evacuation. So that's how we come up with the 36 hour proposal in our criteria.

CHAIRMAN ZECH: Is it fair to say, just to clarify this emergency planning situation, that you don't have traditional offsite emergency planning; do you have any planning at all? Are you proposing any planning at all for emergency planning offsite?

MR. KING: The ingestion pathway planning would be the same as LWR's. The movement of people, people evacuation would only include the early notification.

CHAIRMAN ZECH: Do you have an offsite emergency plan?

MR. KING: Yes, I think there would be something on the shelf that would be called an offsite emergency plan.

COMMISSIONER CARR: According to your paper, you would have one, but it would exclude early notification and detailed evacuation.

CHAIRMAN ZECH: That's what I understand, but you'd have a plan. It would not be what we know it now as a traditional plan.

MR. KING: That's right.

CHAIRMAN ZECH: But you would have a plan.

MR. KING: You would have a plan, but it would not have --

CHAIRMAN ZECH: But it would be quite different as we know it now.

MR. KING: That's right.

CHAIRMAN ZECH: All right.

COMMISSIONER ROGERS: Excuse me. I think we ought to be really very clear on what this is all about. I'm not convinced that I've heard the same story here at all on whether there's going to be a plan or not a plan and whether it's required or not required.

CHAIRMAN ZECH: That's what I'm trying to determine.

COMMISSIONER ROGERS: It certainly wouldn't involve evacuation exercises. That I understand. What it does involve, isn't very clear, because the statement here in the slide simply says that a traditional offsite emergency planning would not have to include -- it says some things it wouldn't have to include. It doesn't say what it would have to include and it doesn't say whatever it does include is essential.

COMMISSIONER CARR: I assumed it included everything but those.

COMMISSIONER ROGERS: It doesn't say so and we just heard from someone in the back of the room that said there doesn't have to be a plan.

CHAIRMAN ZECH: It need clarification. General

Counsel, maybe you want to comment.

MR. PARLER: The way that I understand the situation, which is probably an oversimplification is that there would be for these reactors or this category, offsite emergency planning treated in same sort of the fashion that it was with the TMI-2 accident. We would have certain guidelines in the Appendix E, but if you take all of these things out, they're eliminated on page 27 -- that is, not early notification, not detailed evacuation or provisions for exercising the plan; there would be very little left with the plan.

In my judgment, I would go back to the pre-TMI-2 emergency planning days. It is the licensing appeal board, by the way, that handled one of the first set of issues in the emergency planning area under the old guidelines prior to TMI-2, so that's the basis for my comment.

CHAIRMAN ZECH: You envision something in accordance to the General Counsel's description? Is that what you have in mind?

MR. STELLO: I think you're asking the question. We really don't have the full answer, because what it would take is developing a new appendix for the kind of planning we're talking about. We haven't done that. What we are saying is that the current Appendix E, required for light water reactors, would not be necessary.

CHAIRMAN ZECH: Well, the situation needs

clarification.

MR. STELLO: Absolutely.

CHAIRMAN ZECH: No question about that.

MR. STELLO: The big issue here is we think for the Commission, is a kind of approach that is being taken is one where the kind of emergency planning outlined in Appendix E is not required. Something substantially lesser than that kind of detail, exercise and planning, although we have not prescribed it or attempted to describe what that is yet, would be required.

The issue is, is that a policy issue that the Commission would entertain and it would only be after the Commission decides yes, you would entertain it, that I think it would warrant putting any resources to develop specifically what kind of planning we're talking about.

COMMISSIONER CARR: I think it's fair to say that there would be a plan but we don't know what's in it.

MR. STELLO: And it won't come up to the standards of Appendix E.

COMMISSIONER ZECH: We can't give you any definitive guidance other than yes, we'll take a look at it, but we need to know -- we need to see it clarified. We need to know exactly what you're talking about.

MR. STELLO: In my view, I think you need to see the SERs for these plants too to understand the total context of

these.

COMMISSIONER ZECH: And I think that's very true. On the other hand, we don't want the SERs to go down the pipe so far it give you kind of a false start in the whole thing too. So, we do need to see what you have in mind a little bit more clearly than we've got it here today in this particular issue anyway, I think.

MR. STELLO: I think one of the things we're going to want to do as you already indicated is deal with the comments of the ACRS and have a paper for the Commission that would do that and we would just -- at least try to add a little bit more substance to the answer to the question to what extent would emergency planning be needed and what kind of features would we expect in such plants and provide that to the Commission as well with a schedule that I hope would not interfere with us getting the draft SERs that we're now scheduled to get out in September.

COMMISSIONER ZECH: Well, I think we can go ahead with the draft SERs but I emphasize draft because I think the ACRS has made some very useful comments and I think that they should be addressed. I think their approach as I understand it is to encourage cautious approach and conservative approach which I think certainly I agree with. I hope we all will as we go in this advanced reactor business.

I think we all recognize we want to have timely

action. We want to move ahead but we do need a responsible, disciplined approach to the whole subject so that we can be confident that we're stepping into this area using the experience we've gained from the light water reactor operational design construction experience that we've already gained.

It's awfully important, I think, that we try to do that and so we want to move forward, of course. We want to support these new initiatives, but we also have to feel that we are giving you the guidance that we have confidence in will be appropriate.

Are you finished yet or do you have a couple of more slides to go yet, I think; don't you?

MR. KING: Two more. Page 28 and 29 deal with what we plan to do with these criteria.

COMMISSIONER ZECH: Okay.

[Slide.]

MR. KING: At the conceptual design stage, we're proposing these criteria as something that we would use to evaluate the DOE conceptual designs assuming they're endorsed by the Commission, then factored into the SER and provided the SERs be provided to DOE as preliminary guidance on their designs.

Whether we actually try and formally implement these criteria via rulemaking is a follow on decision we will make

after these conceptual design stages.

ACRS had a comment that we continue development and dialogue in these areas and we agree with that. We have planned with DOE follow up discussions beyond the conceptual design stage to look at R&D program results, design evolution and so forth.

[Slide.]

MR. KING: The next stage for the designers would be to submit a preliminary design and proceed toward a PDA or if they want to go right to a final design and get final design approval. That's several years down the road. We would plan to build upon these criteria and the guidance provided in our SER in reviewing these designs at the PDA or FDA stage.

Then at the design certification stage following completion of prototype testing we would use these criteria, accepted in the verification process, documented in the design certification process. That completes --

MR. STELLO: The ACRS commented I think in this area a concern that we would put designers -- licensees at some significant risk if we postponed promulgating requirements or rules for such designs since when we did that then they went down the road and made change. The whole idea, in my view, is I don't think that licensees, the industry will start buying these things until they're certified.

At the time that they're certified, then that in fact

has the Commission saying that is the design which we accept. I don't think at that point there's any problem.

COMMISSIONER ZECH: Yes, General Counsel, go ahead.

MR. PARLER: May I make a comment?

COMMISSIONER ZECH: Please do.

MR. PARLER: On that, I said in this paper on page 25 that the office that I'm responsible for had no legal objections to the paper. The views also said there were many important issues addressed in the paper which would be resolved by rulemaking well in advance of any design certification, rulemaking. On a number of occasions before this Commission, in the more recent past were plant extensions for the repository in the future and for the standardization generally that we talked about several weeks ago, the SECY 88-169. I emphasize the importance of having at least in my judgment again on the basis of past experience with the LWR licensing -- for having the technical requirement specified in the regulations at least the performance objectives as clearly and as completely as we could before the important decision points were reached otherwise procedurally when the important procedural points are reached, you would have a very difficult time as we have had in the past.

The comment was made at the beginning of this briefing that we will worry later about the existing requirements in the regulations and about changing the

regulations. The point is a very simple one. If you wait to resolve all of that until we get to the design certification stage, there will be many, many procedural problems and if that can be avoided, it should be. That was what was meant by the position of the Office of General Counsel, on page 25 of the staff paper.

COMMISSIONER ZECH: All right. Thank you very much.

MR. STELLO: We're through, Mr. Chairman.

COMMISSIONER ZECH: All right. Questions of my fellow Commissioners? Mr. Roberts?

COMMISSIONER ROBERTS: No question. If DOE is going to build a HTGR at Hanford, I'm suddenly beginning to think that would go a long way toward the prototype testing. That's all I have.

MR. STELLO: It could.

COMMISSIONER ZECH: Mr. Carr?

COMMISSIONER CARR: On the emergency evacuation issue or the planning, we can always tack that on, I mean, you know, that's separate from everything else here. If you don't like what you see, you can have an evacuation plan made. So, I'm not sure how much time it's worth spending at this point trying to decide exactly what I'll put in that plan. I think when they send their paper up, they can say, this is generally what we think is going to be in there and that will be enough for us to work with right now, I would think. That's all I got.

COMMISSIONER ZECH: Commissioner Rogers?

COMMISSIONER ROGERS: I would agree with Commissioner Carr on that. On the other hand, I think the containment question is one that -- that doesn't fall into that category. It seems to me we have to be very clear on that. That's very much a part of the design.

COMMISSIONER CARR: I reiterate conventional containment because there are people who think they've got three containments in that fuel and --

COMMISSIONER ROGERS: Well, I think -- you know that illustrates the point that we all may be talking about something that's a little bit different and I think that a great deal of clarification is probably necessary there on that whole containment issue as to whether there's anything around it at all, whether there's a confinement of some kind, whether there's a traditional containment.

COMMISSIONER CARR: How they're going to contain the fission fragments important?

COMMISSIONER ROGERS: And I think that's one that probably we have to see some options on.

COMMISSIONER ZECH: All right.

MR. STELLO: You mean "options" meaning have a containment or not have one? That decision has already been made, hasn't it?

You know that would be -- that would be, to go back

and ask the question, give me a design with a containment as an option, is an enormous amount of work for DOE.

COMMISSIONER CARR: Well, I think you've got to show how it's going to be contained, no matter whether you have a contained --

MR. STELLO: Oh, that's a different question.

COMMISSIONER CARR: There's various ways of containing it.

COMMISSIONER ROGERS: That's my point. We may be all talking a little bit about something that's a little bit different here, but I think we need some clarification

MR. STELLO: Yes. But my point is, the issue will be in demonstrating how you achieve the equivalent of containment through a different concept other than a conventional containment system. And that's what the designers' kind of approach --

COMMISSIONER ROGERS: Well, there's going to be some kind of a box around this thing, and what is that box going to look like?

MR. STELLO: Yes.

COMMISSIONER ROGERS: What are the options on that box, all right?

MR. STELLO: Okay.

COMMISSIONER ROGERS: Let's put it that way.

MR. STELLO: All right.

CHAIRMAN ZECH: Have we given the consideration to peer review of this subject? Have we given the consideration of a public comment period somewhere or other to involve as much expertise as we can in these decisions?

MR. STELLO: Only after the Commission -- after some sort of a safety evaluation --

CHAIRMAN ZECH: You're thinking on this line.

MR. STELLO: Yes.

CHAIRMAN ZECH: What's the --

MR. STELLO: That would be part of the process after --

COMMISSIONER CARR: Well, we've had a peer review of this paper, though.

MR. KING: Yes. Yes, we've had three outside experts look at it. We've had ACRS look at it. We have not sent any of these items out for the traditional public comment.

CHAIRMAN ZECH: What is your plan on getting public comment? How would that come into the procedure?

MR. STELLO: Well, what we need is some guidance from the Commission on some of these key issues to finally finalize some kind of safety evaluation report that we would then issue, and it would be at that point at which it would be proper to get comment.

CHAIRMAN ZECH: Fine. All right. How about rulemaking then?

MR. STELLO: Well, rulemaking, by definition, those would go on with comment. In fact, we might want to consider some innovative ways to deal with that, because of the potential breadth of that problem and dealing with the procedure question.

MR. PARLER: I think you were talking about the timing for the rulemaking, weren't you, Mr. Chairman?

CHAIRMAN ZECH: Yes, I really was talking about the timing of it. And also I think the General Counsel has made a point that is important, if I understand it correctly, and we should use our experience on trying to set aside as many of the unknowns as we possibly can, so that when we do come to the point of making decisions, we're not making modifications and changes and additional diversions of the design or whatever.

So we need to come up as much as we possibly can with good, firm guidance for you. We recognize that, and I think that we need to do that.

But the containment issue is something that we need to be confident it, you need to be confident in. The emergency planning, I think, needs clarification. There's a number of loose ends, if you will, I think, that we need to focus on.

I think this briefing has been very useful in showing us the complexity of the issue we're dealing with. I know you're focusing on some of the key issues, but I think we need -- I think we need to continue working this problem very

carefully and focus in on the specifics of some of the things that we're talking about, so that we'll all have the confidence that we can move into this new area with the -- using the experience that we've had in the past. I think it's very important that we do that and learn the lessons that we've learned, some of them the hard way.

MR. STELLO: May I comment, Mr. Chairman?

CHAIRMAN ZECH: Sure.

MR. STELLO: We are talking today about three specific advanced reactor concepts of DOE. In the future, there are other advanced reactor concepts, significantly different conceptually than those three we've talked about today.

The issue of rulemaking comes about for all of those kinds of designs. The resource impact on the Staff of dealing with issues such as rulemaking is one I want to be very deliberate about, because it's a very significant resource issue, and if these are indeed concepts which DOE will advocate, for which there is going to be a concept advanced and a real reactor built, then I think we're justified in putting those resources forward to do all those things.

I think we need to be very selective to make sure that we are putting our effort into the rulemaking areas in those areas for which we are reasonably confident that that's the way in which the development of the industry will take

shape, because we know there are at least five different concepts out there and possibly more. Hence the concept of what to deal with in rulemaking, and example being in these cases you deal with containment as a significant issue with respect to rulemaking. In the other concepts that we're talking about, that is not an issue. So dealing with rulemaking and the advanced reactor concepts as here today, that particular issue would not be applicable to others.

MR. STELLO: So I think we'd like the opportunity to examine very carefully how this is going to go before we would entertain developing that kind of commitment and committing those sorts of resources.

MR. PARLER: May I make several comments?

CHAIRMAN ZECH: Please.

MR. PARLER: If indeed the efforts are going to move forward beyond -- to the design certification stage and beyond, if there isn't rulemaking in advance, there would be substantial resource problems, too, because the licensing proceedings will go on and on.

I would assume that if somebody comes even to this agency for a design certification, that perhaps indicates that there is some interest someplace in using the design.

My final point would be in regard to getting comments on these criteria -- that is, short of rulemaking. I would suggest, at least it occurs to me that if one would wait until

the Staff safety evaluations are prepared and then seek comment on the criteria that were used to prepare those safety evaluations, that might not come across too well.

It would seem to me that it would perhaps be better, whatever the Commission gives the Staff, its guidance on these criteria, to consider putting the criteria out, not in the rulemaking proceeding, but just out for public comment, and then do that before we issue the safety evaluation.

CHAIRMAN ZECH: Thank you.

Did you hear the comment General Counsel made?

MR. STELLO: Yes, Mr. Chairman, I did, and I agree with him. If we really are going to have a design and it is to be certified, we want to do everything we can not to burden the process.

However, it isn't clear to me whether any of these designs or all of them or completely different designs will be the ones certified, and it is those to which I think we ought to direct our resources.

COMMISSIONER CARR: But you didn't address his last point, which was once we give you some guidance on the criteria, that we ought to publish the criteria for public comment.

CHAIRMAN ZECH: Before you publish the SERs.

MR. PARLER: That opportunity also might give you some input as to whether these things might be pursued.

But in any event, if they're not going to be pursued, it's kind of puzzling to me why one at this time isn't asking the Commission for guidance on the criteria.

MR. STELLO: Well, these are -- remember with the task we have, these are conceptual designs that have been developed this far from which DOE may very well make a selection of one or none and move forward. If DOE does make a decision that changes and drops one of the designs or whatever --

COMMISSIONER ROBERTS: Well, would there be just one? Do you know that with certainty?

MR. STELLO: No. But I at least -- maybe we need to have this conversation privately. I think there's some information which I think is highly speculative that I need to get to the Commission and maybe pursue -- pursue it further before we pursue this conversation.

COMMISSIONER CARR: Let me ask one more question about the criteria per se.

My understanding is if the designers come forward with their designs, and those designs fit the criteria you've got here, then you would agree with the designers that they would probably not need the containment or they would have this. So these are the criteria you're going to judge those designs by, and that's what you'd like us to agree with you on.

MR. STELLO: Correct.

COMMISSIONER CARR: But should they meet your criteria, it's your feeling that safety is all right and the public health, then.

MR. STELLO: We're suggesting to the Commission that we would not be able to go along with those design concepts unless they follow --

COMMISSIONER CARR: Unless they might this criteria.

MR. STELLO: Those -- that's correct.

COMMISSIONER CARR: But if they do, then public health and safety will be assured.

MR. STELLO: That's correct.

CHAIRMAN ZECH: All right. What I think we need to do is, you get together with General Counsel and kind of work out the process that we're talking about.

MR. STELLO: Okay.

CHAIRMAN ZECH: So that you can look at your responsibilities, and General Counsel can look at his, and that perhaps you can both get together and come to the Commission with something that we can consider a little bit more carefully.

What we're trying to do, of course, is to encourage, you know, advanced reactors that have enhanced safety features. We're trying not to be a bottleneck in our regulatory approach to this whole matter. We want to make sure that we do, though, make the proper decisions, so that -- we don't want to preclude

any advancements in technology if it's in front of us.

Those who are in the design -- reactor design business that believe they really have something to bring forward that will be helpful to our country, will be safe, we want to look at it, but we have to use our process. We want to make sure we're confident that we are doing the right thing, and I would ask that EDO and the General Counsel get together and perhaps present us a joint paper that we can look at as far as the process is concerned, and we may be able to give you guidance along the line that you're asking for after you've done that.

Are there any other comments from my fellow Commissioners?

[No response.]

CHAIRMAN ZECH: All right. Thank you very much. We stand adjourned.

[Whereupon, at 3:40 o'clock, p.m., the Commission meeting was concluded.]

CERTIFICATE OF TRANSCRIBER

This is to certify that the attached events of a meeting of the U.S. Nuclear Regulatory Commission entitled:

TITLE OF MEETING: BRIEFING ON KEY ISSUES ASSOCIATED WITH DOE ADVANCED REACTOR CONCEPTS

PLACE OF MEETING: Washington, D.C.

DATE OF MEETING: TUESDAY, AUGUST 9, 1988

were transcribed by me. I further certify that said transcription is accurate and complete, to the best of my ability, and that the transcript is a true and accurate record of the foregoing events.

Suzanne Young

Ann Riley & Associates, Ltd.

BRIEFING ON

KEY LICENSING ISSUES

ASSOCIATED WITH DOE

ADVANCED REACTOR CONCEPTS

BACKGROUND

- SINCE LATE 1986 NRC HAS HAD UNDER REVIEW THREE DOE SPONSORED ADVANCED REACTOR CONCEPTUAL DESIGNS:

  ° 350 MWT MODULAR HTGR
  ° 425 MWT MODULAR LMR (PRISM)
  ° 900 MWT MODULAR LMR (SAFR)

- STAFF REVIEW PLANS ARE DESCRIBED IN SECY-86-368 (DATED 12/10/86):

  ° COMMISSION PAPERS TO REQUEST COMMISSION GUIDANCE ON MATTERS WITH POLICY IMPLICATIONS.
  ° SERS BEING WRITTEN ON EACH DESIGN

- PURPOSE OF THE STAFF REVIEW IS TO PROVIDE GUIDANCE EARLY IN THE DESIGN PROCESS ON LICENSING REQUIREMENTS, CONSISTENT WITH THE COMMISSION'S ADVANCED REACTOR POLICY STATEMENT.

COMMISSION PAPERS ON ISSUES WITH POLICY IMPLICATIONS

1) Licensing Issues (SECY-88-203):

- Accident Selection
- Siting Source Term Selection and Use
- Adequacy of Containment
- Adequacy of Emergency Planning

2) Standardization Issues (SECY-88-202):

- Scope of design to be certified
- Level of design detail to be certified
- Plant Options to be certified
- Prototype Testing

KEY FEATURES OF DOE PROPOSED DESIGNS

STANDARDIZATION

- MODULAR REACTORS
- GOAL IS DESIGN CERTIFICATION
- CONCENTRATE ALL SAFETY FUNCTIONS IN NUCLEAR ISLAND.

SAFETY

- PASSIVE DECAY HEAT REMOVAL AND REACTOR SHUTDOWN SYSTEMS WHICH HAVE THE POTENTIAL TO PREVENT CORE DAMAGE UNDER SEVERE CHALLENGES, SUCH AS:

    - ATWS
    - STATION BLACKOUT
    - REACTIVITY INSERTION EVENTS
    - LOSS OF FLOW EVENTS
    - LOSS OF HEAT SINK EVENTS

- GREATLY REDUCED NEED FOR OPERATOR ACTION AND REDUCED POTENTIAL FOR HUMAN ERROR TO AFFECT SAFETY SYSTEMS.

- MECHANISTIC SITING SOURCE TERM CALCULATION.

- NO CONVENTIONAL CONTAINMENT BUILDING.

- PROPOSED REDUCTION IN OFFSITE EMERGENCY EVACUATION PLANNING.

OVERALL APPROACH TO REVIEW AND DEVELOPMENT OF GUIDANCE

GUIDANCE FROM POLICY STATEMENT - FROM THE COMMISSION'S ADVANCED REACTOR POLICY STATEMENT: "THE COMMISSION EXPECTS, AS A MINIMUM, AT LEAST THE SAME DEGREE OF PROTECTION OF THE PUBLIC AND THE ENVIRONMENT THAT IS REQUIRED FOR CURRENT GENERATION LWRS. FURTHERMORE, THE COMMISSION EXPECTS THAT ADVANCED REACTORS WILL PROVIDE ENHANCED MARGINS OF SAFETY AND/OR UTILIZE SIMPLIFIED, INHERENT, PASSIVE, OR OTHER INNOVATIVE MEANS TO ACCOMPLISH THEIR SAFETY FUNCTIONS."

STAFF REVIEW APPROACH - APPLICABLE SAFETY GUIDANCE PROMULGATED FOR LWRS USED AS THE BASIS TO DEVELOP THE GUIDANCE FOR ADVANCED REACTORS:

- 10 CFR
- SRP
- SEVERE ACCIDENT POLICY
- SAFETY GOAL POLICY
- STANDARDIZATION POLICY

- THE STAFF DEVELOPED GUIDANCE FOR THE DOE CONCEPTS IS STRUCTURED TO:

    - DEFINE MINIMUM REQUIREMENTS TO ENSURE AT LEAST THE SAME DEGREE OF PROTECTION AS LWRS.

    - ADDRESS EXPECTED ENHANCED SAFETY.

KEY ELEMENTS OF STAFF PROPOSAL

1) AN APPROACH TO MAINTAIN AT LEAST THE SAME DEGREE OF PROTECTION OF THE PUBLIC AND THE ENVIRONMENT AT LWRS BY:

- EVALUATING THE VARIOUS FACTORS THAT CONTRIBUTE TO SAFETY ON AN LWR AND REQUIRING THE ADVANCED DESIGNS HAVE THE SAME FACTORS, OR FACTORS THAT ARE JUDGED TO PROVIDE THE SAME DEGREE OF SAFETY (DEFENSE IN DEPTH).

- NOT RELYING ON A SINGLE MEASURE OF SAFETY, SUCH AS A COMPARISON OF PRA RESULTS OR SAFETY GOAL COMPLIANCE.

2) VERIFICATION OF THE UNIQUE PLANT SAFETY CHARACTERISTICS VIA PROTOTYPE TESTING OF THE PLANT RESPONSE TO VARIOUS CHALLENGING EVENTS:

- DESIGNS WITH LOW POTENTIAL FOR CORE DAMAGE CAN BE TESTED TO MORE CHALLENGING EVENTS.

- FOR THOSE SEVERE CHALLENGES INCLUDED IN THE TEST PROGRAM, THIS PROVIDES AN OPPORTUNITY TO ELIMINATE UNCERTAINTY IN THE PLANT RESPONSE BEYOND WHAT HAS BEEN DONE OR MAY BE ACHIEVABLE ON CURRENT GENERATION LWRs.

- EXAMPLES OF SEVERE CHALLENGES TO BE INCLUDED IN TEST PROGRAM:

  - ATWS EVENTS
  - STATION BLACKOUT EVENTS
  - LOSS OF HEAT SINK EVENTS

EXAMPLES OF STAFF APPROACH TO ASSURING AT LEAST THE SAME DEGREE OF PROTECTION AS LWRS

FACTORS CONTRIBUTING TO SAFETY	LWRS	DOE CONCEPTS
ACCIDENT PREVENTION	- ACCEPTED AND CONSERVATIVE DESIGN CODES AND PRACTICES.	- USE APPLICABLE LWR CRITERIA/STANDARDS. DEVELOP ADDITIONAL CRITERIA/STDS, AS NECESSARY.
	- QUALITY DESIGN, CONSTRUCTION OPERATION, MAINTENANCE	- SAME, WITH POSSIBLE ENHANCEMENT IN CERTAIN AREAS.

FACTORS CONTRIBUTING TO SAFETY	LWRS	DOE CONCEPTS
PROTECTION/MITIGATION	-REQUIREMENTS ON PERFORMING KEY SAFETY FUNCTIONS:	
	° REACTOR SHUTDOWN	- SAME
	° DECAY HEAT REMOVAL	- SAME
	° CONTAINMENT BLDG.	- (*) SUBSTITUTE HIGH DEGREE OF CORE DAMAGE PREVENTION (VERIFIED BY PROTOTYPE TESTING) FOR CONTAINMENT BLDG.

(*) KEY POLICY ISSUES DERIVE FROM DIFFERENCES IN DOE APPROACH AND STAFF PROPOSALS FROM THAT USED IN THE TRADITIONAL LWR SAFETY APPROACH.

FACTORS CONTRIBUTING TO SAFETY	LWRS	DOE CONCEPTS
	- LIMITS ON CORE DAMAGE FROM DBAS AND CONTAINMENT FAILURE FROM SEVERE CHALLENGES.	- LIMITS ON CORE DAMAGE FROM DBAS + SEVERE CHALLENGES.
	- LIMITS ON FP RELEASE FROM VARIOUS EVENT CATEGORIES	- SAME

Factors Contributing to Safety	LWRs	DOE Concepts
Safety Analysis, Documentation and Limits	- Selection of appropriate DBAs to be considered in the design plus analysis on accommodation of severe challenges.	- Same, with severe challenges considered in the design for siting determination. *
	- Thorough safety analysis:	- Verified by prototype testing
	° DBAs-conservative analysis	° Same
	° Severe challenges best-est. analysis	° Same
	° PRA	° Same
	- Tech. Specs.	- Same
	- Surveillance & testing programs	- Same

(*) Key policy issues derive from differences in DOE approach and staff proposals from that used in the traditional LWR safety approach.

Factors Contributing to Safety	LWRs	DOE Concepts
Siting	- Deterministic source term based upon core melt accid. Offsite release analysis assumes containment integrity. 10CFR100 determination.	- (*) Source term and release based upon a mechanistic analysis of a range of severe accidents. 10CFR100 dose determination.
Emergency Planning	- Preplanned offsite evacuation required.	- (*) "Ad hoc" evacuation acceptable if sufficient time available.

(*) Key policy issues derive from differences in DOE approach and staff proposals from that used in the traditional LWR safety approach.

	- PREPLANNED INGESTION PATHWAY ACTIONS REQUIRED.	- SAME
OPERATING EXPERIENCE	- MUCH AVAILABLE AND MANY LESSONS LEARNED.	- SOME LWR EXPERIENCE APPLICABLE. PROTOTYPE TEST WILL ADD TO EXPERIENCE.

Factors Contributing to Safety	LWRS	DOE Concepts
Human Factors	- Training	- Same
	- Licensed Operators	- Same
	- Safety Grade Control Room	- Simplified passive systems and reduced vulnerability to human error compensate for control room and operator function req'ts.
Sabotage	- 10CFR73 requirements	Same. Underground location and passive safety features evaluated for protection versus LWR features.

PROTOTYPE REACTOR TEST

- PROTOTYPE REACTOR:
  - ONE MODULE, ASSOCIATED INSTRUMENTATION AND CONTROLS AND OTHER SYSTEMS IMPORTANT TO SAFETY AT LEAST THROUGH AND INCLUDING THE STEAM GENERATOR,
  - BUILT TO SAME STANDARDS AND DESIGN AS THE PLANT TO BE CERTIFIED,
  - IF PROTOTYPE DOES NOT INCLUDE THE WHOLE PLANT, MUST SIMULATE INTERFACE REQUIREMENTS
  - SOME SPECIAL INSTRUMENTATION AND/OR TEST FEATURES MAY BE REQUIRED.

° TEST PROGRAM:

- PURPOSE IS TO VERIFY PLANT RESPONSE TO BOUNDING EVENTS AND GENERATE SUFFICIENT EXPERIMENTAL DATA TO VERIFY ANALYTICAL TOOLS.

- SHOULD BE CONDUCTED IN A STEPWISE FASHION FROM LOW POWER/LOW DECAY HEAT CONDITIONS TO HIGHER POWER/HIGHER DECAY HEAT CONDITIONS AND FROM FRESH CORE TO EQUILIBRIUM CORE CONDITIONS,

- DIRECTED TOWARD INTERNAL EVENTS,

- TESTING THAT DAMAGES PLANT IS NOT NECESSARY,

- DOE SHOULD PROPOSE DETAILED TEST PROGRAM, AT A LATER REVIEW STAGE, FOR NRC REVIEW AND APPROVAL.

- DESIGN CERTIFICATION DEPENDENT UPON TEST RESULTS.

Example Test Matrix (MHTGR)

Major Parameters of Interest*	BOL	MOC	EOC	Equilibrium Core Burnup
1) Negative Temperature Coefficient (ATWS)	X	X	X	X
2) Passive decay heat removal under LOF/LOHS/SB conditions:	X			X
- pressurized (fuel temp/RV temp)	X			X
- depressurized (fuel temp/RV) temp)	X			
- conduction to ground (fuel temp/RV temp/cavity temp)	X			

Major Parameters of Interest*	BOL	MOC	EOC	Equilibrium Core Burnup
3) Fission product holdup/ plateout in reactor cavity	X			
4) Performance under various system configurations:	X			
- Various trains of active heat removal system	X			
- Partial blockage of passive heat removal system	X			
- Etc.				

---

* Prototype test will also verify many other important design/safety factors.

APPROACH TO ADDRESSING EXPECTED ENHANCED SAFETY

- DESIGNER SHOULD DOCUMENT ENHANCED SAFETY CHARACTERISTICS/MARGINS
- IMPLEMENT ADDITIONAL ENHANCEMENTS IF:
    1) MARGINS ARE SMALL
    2) JUSTIFIED ON A COST/BENEFIT BASIS

STAFF PROPOSED CRITERIA FOR RESOLUTION OF FOUR KEY ISSUES

ACCIDENT SELECTION

ESTABLISH FOUR CATEGORIES OF EVENTS WHICH MUST BE CONSIDERED AS DEFINED BELOW:

1) EVENT CATEGORY I (EC-I)

- EQUIVALENT TO ANTICIPATED OPERATIONAL OCCURRENCES.
- SELECTED VIA ENGINEERING JUDGEMENT, COMPLEMENTED BY PRA, AND GENERALLY INCLUDES EVENTS WITH A FREQUENCY DOWN TO APPROXIMATELY 10^{-2}/YR.
- USED FOR ESTABLISHING COMPLIANCE WITH 40CFR190 AND 10CFR50, APPENDIX I.

2) EVENT CATEGORY II (EC-II)

- EQUIVALENT TO POSTULATED ACCIDENTS/DBAS.
- SELECTED VIA ENGINEERING JUDGEMENT, COMPLEMENTED BY PRA, AND GENERALLY INCLUDES EVENTS WITH A FREQUENCY DOWN TO APPROXIMATELY 10^{-4}/YR PLUS SOME TRADITIONAL EVENTS.
- USED IN SITING DETERMINATION AS DESCRIBED LATER.
- CONSERVATIVE ANALYSIS (SINGLE FAILURE CRITERIA, NO CREDIT FOR NON-SAFETY GRADE EQUIPMENT, ETC.).

3) EVENT CATEGORY III (EC-III)

- EQUIVALENT TO THE RANGE OF SEVERE ACCIDENTS BEYOND THE TRADITIONAL DBAS WHICH SHOULD BE CONSIDERED IN THE DESIGN CONSISTENT WITH THE COMMISSION'S SEVERE ACCIDENT AND SAFETY GOAL POLICY STATEMENTS, COMPENSATES FOR DETERMINISTIC SITING SOURCE TERM.

- SELECTED VIA ENGINEERING JUDGEMENT, COMPLEMENTED BY PRA AND INCLUDES:

  - INTERNAL EVENTS WITH A FREQUENCY DOWN TO APPROXIMATELY 10^{-7}/YR.
  - EXTERNAL EVENTS CONSISTENT WITH WHAT IS TO BE APPLIED TO LWRS AS PART OF SEVERE ACCIDENT POLICY IMPLEMENTATION.
  - BOUNDING EVENTS SELECTED BY ENGINEERING JUDGEMENT TO COVER UNCERTAINTIES

- USED IN SITING DETERMINATION
- BEST ESTIMATE ANALYSIS.

4) EVENT CATEGORY IV (EC-IV)

- USED IN ASSESSMENT OF EMERGENCY PLANNING NEEDS.
- INCLUDES INTERNAL EVENTS OF SIMILAR FREQUENCY TO THOSE CONSIDERED IN ASSESSING LWR EMERGENCY PLANNING NEEDS (NUREG-0396)

SITING SOURCE TERM AND USE

- TO ALLOW THE USE OF MECHANISTIC ANALYSIS FOR SITING SOURCE TERM SELECTION, THE FOLLOWING EVENT CATEGORIES AND RELEASE GUIDELINES WOULD APPLY FOR SITING DETERMINATIONS:

EVENT CATEGORY	DOSE GUIDELINES	METEOROLOGY
EC-II	10% OF 10CFR100	CONSERVATIVE
EC-III	10CFR100	CONSERVATIVE

- ENSURE NONE OF THE EC-II AND EC-III EVENTS ARE ON A THRESHOLD WHERE A SLIGHT CHANGE IN ASSUMPTIONS CAN CAUSE AN UNACCEPTABLE CHANGE IN SOURCE.

ADEQUACY OF CONTAINMENT

- FOR ACCEPTANCE OF A DESIGN WITHOUT A CONVENTIONAL CONTAINMENT BUILDING, THE FOLLOWING CRITERIA SHOULD BE MET:

  ° PROVIDE MULTIPLE BARRIERS TO RADIATION RELEASE WHICH MEET THE RELEASE GUIDELINES FOR EVENT CATEGORIES I THROUGH III.

  ° DEMONSTRATE ACCEPTABLE PLANT PERFORMANCE OVER A RANGE OF SEVERE CHALLENGES, VIA TESTING UTILIZING A REMOTELY SITED, FULL SIZE PROTOTYPE REACTOR,

  ° PROVIDE ADDITIONAL OR ENHANCED QA, SURVEILLANCE, IN-SERVICE INSPECTION/TESTING, AS NECESSARY, TO ENSURE THAT THE SYSTEMS, STRUCTURES AND COMPONENTS WHICH CONTRIBUTE TO PERFORMING THE CONTAINMENT FUNCTION ARE, IN FACT, CAPABLE OF PERFORMING THEIR FUNCTION OVER THE LIFE OF THE PLANT.

  ° PROVIDE PROTECTION OF SAFETY RELATED SYSTEMS, STRUCTURES AND COMPONENTS FROM SABOTAGE AND EXTERNAL EVENTS EQUIVALENT TO THAT FOR LWRS.

- ELIMINATE CORE MELT, SIGNIFICANT POSITIVE REACTIVITY FEEDBACK OR OTHER ACCIDENTS WITH THE POTENTIAL OF A LARGE RADIATION RELEASE FROM THE EC-I, II AND III CATEGORIES.

- ASSESS WHETHER ADDING A CONVENTIONAL CONTAINMENT BUILDING RESULTS IN A SIGNIFICANT IMPROVEMENT IN SAFETY AND IS JUSTIFIED BASED UPON THE CHANGE IN RISK AND COST.

ADEQUACY OF OFFSITE EMERGENCY PLANNING

TRADITONAL OFFSITE EMERGENCY PLANNING WOULD NOT HAVE TO INCLUDE EARLY NOTIFICATION, DETAILED EVACUATION OR PROVISIONS FOR EXERCISING THE PLAN, PROVIDED THE FOLLOWING ARE MET:

- THE LOWER LEVEL PAGS ARE NOT PREDICTED TO BE EXCEEDED AT THE SITE BOUNDARY DURING THE FIRST 36 HOURS FOLLOWING ANY EVENT IN CATEGORIES EC-I, II AND III.

- A PRA ANALYSIS, WHICH INCLUDES EVENTS IN AT LEAST CATEGORIES EC-I THOUGH IV, INDICATES THAT THE CUMULATIVE FREQUENCY OF EXCEEDING THE LOWER LEVEL PAGS AT THE SITE BOUNDARY WITHIN THE FIRST 36 HOURS DOES NOT EXCEED APPROXIMATELY 10-6/YR.

APPLICATION OF CRITERIA

CONCEPTUAL DESIGN STAGE

- CRITERIA, AS ENDORSED BY THE COMMISSION, USED BY THE STAFF IN FINALIZING THE SERs AND PROVIDED TO DOE AS GUIDANCE.

- STAFF REVIEW OF CONCEPTUAL DESIGNS ASSESSES POTENTIAL OF THE DESIGNS TO MEET THE CRITERIA AND ASSESSES ENHANCED SAFETY. RESULTS DOCUMENTED IN SERs.

- RECOMMENDATION ON WHETHER TO FORMALLY IMPLEMENT CRITERIA VIA RULEMAKING TO BE MADE FOLLOWING CONCEPTUAL DESIGN REVIEW.

PRELIMINARY/FINAL DESIGN STAGE

- STAFF REVIEWS PRELIMINARY/FINAL DESIGNS FOR COMPLIANCE WITH CRITERIA, WITH DUE CONSIDERATION OF IMPACT OF HAVING MORE DESIGN DETAIL AND SUPPORTING R&D AVAILABLE (I.E., RECONSIDER DESIGN SPECIFIC ASPECTS/CONSERVATISMS).

DESIGN CERTIFICATION

- STAFF CERTIFIES DESIGN IN COMPLIANCE WITH CRITERIA, VIA RULEMAKING, FOLLOWING COMPLETION OF PROTOTYPE TESTING.

POLICY ISSUE
(Notation Vote)

July 15, 1988 SECY-88-203

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: KEY LICENSING ISSUES ASSOCIATED WITH DOE SPONSORED ADVANCED REACTOR DESIGNS

Purpose: To obtain Commission guidance on the criteria proposed by the staff to address the key licensing issues resulting from the review of the Department of Energy (DOE) sponsored advanced reactor designs. This paper responds to the Chairman's memorandum dated July 9, 1987.

Summary: This paper presents a set of criteria which the staff proposes to use to assess the DOE sponsored advanced reactor concepts in the areas of:

- accident selection
- siting source term calculation and use
- adequacy of containment system
- adequacy of offsite emergency planning

It should be noted that criteria proposed by the staff in this paper are consistent with the criteria proposed in the staff's paper on "Standardization of Advanced Reactor Designs." Specifically, note that Section II.B.3 of this paper and Section C of the Standardization Paper both discuss the need for prototype testing in regard to certifying a design without a containment building.

Due to the unique characteristics of the DOE advanced concepts their approach in the above areas does not follow that of conventional Light Water Reactors (LWRs). The major differences in approach can be summarized as (1) desiring to use a more mechanistic approach in the selection of accidents to be considered in the design and in the calculation of source terms, (2) not including a conventional containment building in the design and (3) elimination of the need for offsite emergency evacuation and drills. Each of these major differences results

CONTACT: Thomas L. King, RES
49-23765

from the characteristics of the designs which, due to their use of passive reactor shutdown and decay heat removal systems, are claimed by DOE to prevent fuel damage for a wide range of accident conditions, including very unlikely events such as anticipated transients without scram, station blackout and multiple operator errors. Accordingly, the staff has looked at the fundamental technical issues associated with each of these areas and has developed an approach and criteria to address each. The approach utilizes the guidance in the Commission's Advanced Reactor, Safety Goal, Severe Accident, and Standardization Policy Statements as the basis for deriving a set of decision criteria against which each advanced concept could be reviewed in the above areas. Consistent with the guidance in the Advanced Reactor Policy Statement, the overall goal of the approach and the proposed criteria is to ensure that advanced reactors achieve a level of safety at least equivalent to that of current generation LWRs (where current generation LWRs are defined as those evolutionary LWRs (ABWR/APWR) currently undergoing review for Final Design Approval). This is achieved by utilizing guidance developed for LWRs, where applicable, and by developing criteria for the unique aspects of the designs which require the application of at least equivalent conservative and quality design, construction and operation standards as LWRs, equivalent radiation release limits, consideration of at least equivalent ranges of events (both internal and external) and demonstrated performance to compensate for the lack of experience as compared to LWRs. It is important to note that the proposed criteria allow a trade off between plant protection and accident mitigation to achieve an equivalent level of safety as LWRs; however, they do not allow elimination of either plant protection or mitigation. In addition to addressing an equivalent level of safety as LWRs, the proposed criteria require each design to be evaluated for expected enhancements in reactor safety, as discussed in the Advanced Reactor Policy Statement, and to make improvement when justified on a cost/benefit basis. The staff expects each new generation of reactor designs to have improved safety characteristics over previous generations.

The proposed criteria are described in Section II of this paper and are structured into a set of "General Criteria" which describe the overall approach and principles being used by the staff in conducting the review of the DOE advanced concepts (these general criteria could also be applied in the review of any advanced reactor design significantly different than current generation LWRs) and a set of "Specific Criteria" which implement the general criteria in the four key areas of accident selection, siting source term, containment and offsite emergency planning associated with the DOE proposed designs.

In summary, the staff criteria addressing the four key areas propose (1) a range of events which must be used for design and for siting purposes, (2) a mechanistic approach for calculating siting source terms, including siting dose limits for the mechanistic siting source terms, (3) criteria that a design without a containment building must meet, including requiring demonstration of its safety via prototype testing prior to design certification and (4) the criteria which would have to be met to accept DOE's proposal to eliminate the requirement for preplanned offsite emergency evacuation drills. Sections II.B.1 through II.B.4 of this paper describe, respectively, the staff proposed criteria in each of the above areas.

If endorsed by the Commission, the criteria proposed in this paper would be used by the staff in preparing preliminary guidance to DOE on the acceptability of their three proposed advanced reactor conceptual designs and supporting R&D and plans leading to design certification. Since this guidance would be preliminary in nature and would not have undergone the formal public process associated with rules and other types of licensing guidance, it is the staff's intent to provide, after the pre-application reviews of the three DOE conceptual designs are complete, a recommendation on the need for rulemaking to put in place a licensing framework for advanced reactors, building upon the approach, criteria and principles discussed in this paper. This would then have the effect of formalizing the work done at the conceptual design review stage and contribute toward stabilizing those licensing requirements for reactor designs significantly different than current generation LWRs. In the interim, in reviewing any actual application for these advanced designs, it would be our plan to have the designers and NRC assess the design for conformance with these criteria at each stage of the review (Preliminary Design Approval/Final Design Approval/Design Certification). An advanced reactor design would not be certified until there exists reasonable assurance that the design was in conformance with the staff's criteria.

<u>Background</u>: In SECY-86-368, "NRC Activities Related to the Commission's Policy on the Regulation of Advanced Nuclear Power Plants," dated December 10, 1986, the Commission was informed of the staff's plan for the review of three advanced reactor conceptual designs sponsored by the DOE. The reactors under review are:

1. A 350 MWt Modular High Temperature Gas-Cooled Reactor (MHTGR);

2. A 425 MWt liquid sodium cooled Power Reactor Inherently Safe Module (PRISM); and

3. A 900 MWt liquid sodium cooled Sodium Advanced Fast Reactor (SAFR).

For all three reactor concepts, DOE has proposed:

- Use of a more mechanistic approach to accident selection and calculation of source terms than used for LWRs;

- A containment system that is different from those on the current generation of LWRs; and

- Elimination of the need for offsite emergency evacuation, sheltering and drills.

As a result of these proposals, the staff has identified certain key licensing issues which, due to their policy implications, require Commission review and guidance prior to the staff finalizing its SER reviews and finalizing the licensing guidance for each of the three advanced concepts. These key licensing issues can be summarized as follows:

1. What range of accidents must be considered for design, siting and emergency planning?

2. How should source terms be calculated and used for designs significantly different than current generation LWRs?

3. How should the adequacy of or need for a containment building be evaluated?

4. How should the adequacy of or need for offsite emergency evacuation, sheltering and drills be evaluated?

The staff approach and proposed criteria to address these issues are discussed below. Due to the similarities in the design approach and key issues among the three DOE advanced designs, the proposed criteria are considered to be generic in nature, although provision has been made in the criteria to address plant specific differences.

It should be noted that the use of more highly enriched uranium (~ 20%) in the MHTGR and plutonium in the LMRs raise policy issues regarding safeguards and fuel cycle that are still pending from previous efforts to close the fuel cycle. However, these issues have not been included in the current staff review (which has been limited to reactor design), but they will be addressed separately in follow-on activities, subsequent to completion of the staff review of the designs.

Discussion: A short summary of the proposed designs is presented for information in Enclosure 1. The discussion that follows is organized to address first the overall approach the staff has taken in the review of the advanced concepts and second the criteria proposed for application in the review.

I) Overall Approach

The Advanced Reactor Policy states that advanced reactors must, as a minimum, provide at least the same degree of protection of the public and the environment that is required for current generation LWRs. In this regard, the staff has interpreted current generation LWRs to be those evolutionary designs currently under review as standard plant designs (ABWR/APWR). Further, the policy states that the Commission expects advanced designs to provide enhanced margins of safety. Accordingly, in the review of the DOE advanced designs, the staff proposes to use and build upon applicable existing regulations and guidelines for safety developed for application to LWRs, to develop additional criteria, when necessary, to address the unique characteristics of these designs, and to require they be assessed for enhanced safety. Further, the staff expects that each new generation of reactor designs will have improved safety characteristics over previous generations. That is, the large advanced LWRs (ALWRs) currently under review (ABWR/APWR) are expected to be better in safety and performance than existing LWRs, the mid-size ALWRs currently under development and targeted for the mid-1990s are expected to be better than the large ALWRs currently under review and finally, the generation of plants discussed in this paper are expected to be better than the previous generations of LWRs.

The existing regulations and guidelines used in the development of these criteria are:

- Applicable portions of 10CFR and LWR Standard Review Plan (NUREG -0800)
- Severe Accident Policy Statement
- Safety Goal Policy Statement
- Standardization Policy

In the application of the above, the staff has, in some cases, had to interpret the guidance developed for LWRs for application to the non-LWR concepts and issues under review. In making such interpretations, the staff has taken an approach directed toward maintaining at least equivalent limits and criteria as LWRs for quality design, construction and operation, for the release of radiation, to maintain defense in depth, to allow provisions for conservatism to account for plant specific uncertainties in the designs, and to be consistent with the guidance under development for future LWRs for the treatment of severe accidents.

Each of these considerations appear in the criteria discussed in Section II of this paper. However, because of the fundamental

importance of the defense in depth principle to reactor safety, it is essential that its application for advanced reactors be specifically addressed. Accordingly, a discussion on defense in depth is provided below.

Defense in Depth

Defense in depth in nuclear power plant safety regulation is a philosophy that entails the use of various layers of requirements that help ensure safety is achieved through multiple, diverse and complimentary means. These layers of requirements address the different stages and aspects of plant safety which can be generally categorized as prevention, protection, mitigation and emergency planning, and include items such as:

1) Plant design using conservative assumptions, appropriate codes and standards and quality in the design, construction, operation and maintenance to minimize the potential for accidents;

2) High reliability, redundancy and/or diversity in components, systems and structures to adequately respond to and protect the plant and the barriers to radiation release in the event of an accident;

3) Mitigative capability to delay and limit the release of fission products to the environment in the event an accident leads to the failure of one or more barriers to radiation release; and

4) Emergency planning for protecting the public in the event radiation release from the plant exceeds acceptable limits.

Enclosure 2 provides a more detailed breakdown of the major components of defense in depth along with an indication of which of these components are utilized by the advanced designs to a greater degree than used by current generation LWRs and also which components are used to a lesser degree. In general, the three DOE advanced designs have attempted to maintain defense in depth by addressing all four categories listed above. However, the advanced reactor designers have approached plant design and the means of maintaining defense in depth somewhat differently than the approach taken by LWR designers. In general, the advanced designs make a shift in emphasis from mitigation features to highly reliable protection features. For example, advanced reactor designers aim to achieve high reliability and protection through the use of simple and passive decay heat removal and reactor shutdown methods as compared to high reliability through active systems in LWR designs (based on LWR

operating experience, passive and simple system designs are encouraged in the Commission's Advanced Reactor Policy Statement). These protection features are directed toward maintaining fuel integrity, even under very unlikely events. Mitigation is provided in the advanced reactor designs through different containment systems, through physical phenomena (fission product retention, plateout and holdup), and through use of the long time response of the reactor in accident sequences. This has resulted in designs which propose to accomplish protection, mitigation and emergency planning in ways different than LWRs, thus raising the issues which are the subject of this paper.

In the development of the criteria discussed in the remaining part of this paper, requirements have been included to ensure that each of the four categories of defense in depth listed in Enclosure 2 is addressed by the advanced designs consistent with their unique characteristics, but with the objective of providing at least equivalent protection to the public when the defense in depth provisions are considered as a whole. In summary, the criteria directed toward the accident prevention aspects of defense in depth for advanced reactors are intended to require at least equivalent accident prevention capabilities as are required for LWRs. The criteria directed toward the protection and mitigation aspects of defense in depth are intended to provide equivalent protection to the public and environment against the release of radiation as LWRs, when viewed together (i.e. some trade-off between protection and mitigation is allowed such as the use of highly reliable passive plant protection features versus a traditional containment building). The criteria directed toward emergency planning are intended to provide an equivalent level of protection in consideration of the characteristics of the advanced designs.

It should be noted that the staff proposed criteria include requirements for independent and diverse means to accomplish the main safety functions (reactor shutdown and decay heat removal) and multiple barriers to prevent the release of radioactive material. It is the staff's judgement that reliance on a single system or plant feature to accomplish these important safety functions (even a highly reliable passive system) is not justifiable in light of the importance of these functions to the protection of public health and safety and in view of the difficulty of predicting the failure mode possibilities in a unique design.

II) <u>Criteria Development</u>

In developing the criteria proposed for use in assessing the key issues, a set of general criteria have been developed which describe the approach and framework applied by the staff in the

review of the DOE concepts, and which could be applied in the review of any reactor significantly different than current generation LWRs. These general criteria are discussed first. Second, specific criteria were developed to implement the general criteria for each of the four key issues associated with the DOE concepts. It should be emphasized that the proposed criteria were developed based upon technical considerations only and are directed toward ensuring an equivalent level of safety as current generation LWRs as well as requiring the designs be evaluated for cost effective safety enhancements. In addition, it should be noted that the criteria were developed in consideration of the long range goal of DOE to certify the advanced reactor designs. Since design certification is their ultimate goal and the plans and supporting R&D proposed by DOE for these designs are directed toward certification, the staff's proposed criteria have been developed from the perspective of what may be required to support design certification.

In implementing the criteria, there is latitude to apply conservatisms to account for design specific uncertainties; however, in consideration of the unique characteristics of advanced designs, the staff has not chosen to mandate that these designs include non-mechanistically chosen large radioactive releases from the core, containment buildings or traditional offsite emergency planning as a way of treating uncertainties. It is recognized that the Commission may view designs that utilize non-mechanistically chosen large radioactive releases and include containment buildings and traditional offsite emergency planning as taking prudent measures to account for uncertainties and to reduce risk. In addition, the reduced operating experience with reactors of this type, as compared to LWRs, may also be viewed as contributing to greater uncertainty and thus to the need for a more conventional approach to specifying release, containment and emergency planning criteria. However, it must be kept in mind that many of the lessons learned as a result of LWR operating experience are applicable to the advanced designs and that the effectiveness of the traditional approaches to specifying release, containment and emergency planning criteria are also dependent upon engineering analysis and judgement.

Therefore, the staff believes that with proper engineering analyses, judgement and demonstrated performance, it is possible for advanced reactor designs to properly account for uncertainties and, as a minimum, achieve a level of safety equivalent to that of current generation LWRs without utilizing a non-mechanistic large release, containment building or traditional offsite emergency planning. In addition, the staff is concerned that deterministically imposing such measures on a reactor design may have a negative impact on other safety characteristics of the design (such as the passive decay heat

removal via the natural convection of air) or may not give proper credit to the innovative safety features of the new designs. Accordingly, the staff proposed criteria define those conditions under which a design would be acceptable without requiring a non-mechanistically chosen large radioactive release, containment building or traditional offsite emergency planning.

II.A) General Criteria

The following general criteria represent a framework and approach for guiding the staff review of the DOE advanced reactor concepts. It is from these general criteria that the specific criteria to address the key issues of accident selection, source term, containment and emergency planning have been derived.

The general criteria proposed are a combination of:

- Criteria which must be met to ensure at least an equivalent level of safety as LWRs (this level of safety is considered as adequate protection),
- Criteria associated with enhanced safety.

II.A.1) Criteria Directed Toward Ensuring at least an Equivalent Level of Safety as LWRs:

i) In the design and review of the advanced reactor concepts, the designers and staff shall utilize applicable existing rules and regulations, as interpreted for advanced reactor concepts (this would involve a process similar to what was done in the review of the Clinch River Breeder Reactor, whereby the LWR Standard Review Plan, General Design Criteria and other regulations were reviewed for their applicability and revised and supplemented, as necessary, to account for the differences and unique attributes of the design as compared to LWRs). The staff's Safety Evaluation Reports on each of the advanced designs will document the use of existing rules and regulations and their interpretation. The following major exceptions to existing rules and regulations are proposed:

- Permit calculation of siting source term based upon mechanistic analysis in lieu of the large non-mechanistic source term applied to LWRs (i.e. the TID-14844 source term used in the 10CFR100 siting determination).

- Permit the containment function to be performed in a fashion different than for LWRs.

- Permit offsite emergency planning to be modified to reflect plant safety characteristics.

(Specific criteria developed for substitution in these three areas are discussed in Sections II.B.2, II.B.3, and II.B.4, respectively).

ii) The advanced reactor concepts shall comply with the intent of the severe accident requirements, which are presently being formulated for LWRs, as follows:

- Meet the four procedural criteria for new plants stated in the Commission's Severe Accident Policy Statement.

- Identify important severe events to be considered in the design (design dependent).

- Evaluate design features incorporated to prevent severe accidents (design dependent).

- Evaluate design features provided for mitigation and accident management (design dependent).

iii) The advanced reactor concepts must show fission product (FP) retention capability at least equivalent to LWRs (i.e. for equivalent classes of events, criteria associated with FP release (fuel damage limits, primary system integrity and offsite dose) from advanced reactors should require the same or better fission product retention than for LWRs.)

iv) The advanced reactor concepts shall maintain the "defense in depth" concept; however, in its application, consideration may be given to the unique safety characteristics of the advanced plants. Some trade-off between prevention and mitigation is acceptable. "Defense in Depth" in performing key safety functions must be maintained equivalent to LWRs via requiring:

- Two diverse, independent means of reactor shutdown, each of which is capable of shutting down the reactor assuming a single failure of active components and without dependence on support systems (electric power, instrument air, etc.). One of the systems must be capable of bringing the plant to cold shutdown indefinitely.

The other system must be capable of bringing the plant to hot shutdown for an extended period of time.

- Two diverse, independent means of decay heat removal, each of which is capable of removing decay heat assuming a single failure of active components.

- Multiple barriers to fission product release.

v) To account for the reduced experience, as compared to LWRs, designs which utilize new or innovative features to perform their safety functions must:

- Demonstrate prior to design certification, via testing on the first of a kind or prototype plant, that reasonable assurance will exist on the ability of these features to prevent or accommodate accidents. Specifics of plant testing can be determined on a case-by-case basis (based upon review of the plant specific safety analysis, Probabilistic Risk Assessment (PRA), etc.) but generally should include sufficient scope of testing on a full size reactor module to demonstrate the performance of the new/innovative safety features over the range of accidents which must be considered in the design. There is not an expectation to demonstrate, by test, all exceedingly rare accident events even though the staff may have required these to be accounted for within the licensing design basis.

- Develop additional QA, inspection, surveillance, and in-service testing techniques and programs, as necessary, to ensure the quality and performance of the new/innovative safety features is maintained within acceptable limits over the life of the plant.

II.A.2 Criteria Associated With Assessment of Enhanced Safety:

i) Applicants must assess and document enhanced safety characteristics/margins such as:

- Long response time
- Reduced potential for operator error
- Capability to retain FP
- Highly reliable safety systems (passive/inherent characteristics)

- Simplification (systems/analysis)

ii) Potential improvements in safety are to be considered when the margins are small or when large improvements in safety can be realized with reasonable cost.

iii) Where enhanced safety/margins are used to reduce uncertainty or to affect the design/operation of the facility, these enhancements must be demonstrated via testing on the first of a kind or prototype plant. Specifics of plant testing can be determined on a case-by-case basis.

II.B) Specific Criteria

Within the framework of the general criteria above, more specific criteria, which implement the above, are provided for each of the four key licensing issues. These specific criteria are discussed next. Additionally, the policy considerations associated with each issue are identified. In developing these criteria, options for addressing each issue, including the pros and cons associated with each option, were evaluated. However, for the purpose of brevity, only the final staff recommendation is provided in this paper.

II.B.1) Accident selection

Issue

What range of accidents need to be considered for advanced reactors to provide a basis for selecting a mechanistic siting source term and for judging the adequacy of containment and offsite emergency planning?

Staff Recommendation:

Selection of a spectrum of accidents which must be considered in the design, beyond the traditional LWR design basis accident (DBA) envelope, is considered necessary for advanced reactors. Consideration of such a spectrum of accidents will: (a) ensure advanced designs comply with the Commission's Safety Goal and Severe Accident Policies, (b) provide a sufficient test of the capability of the design to allow use of mechanistic source terms for siting determinations and for decisions regarding containment and emergency evacuation plans, and (c) ensure the shift in emphasis in defense in depth from accident mitigation to accident protection, as compared to LWRs, does in fact provide designs with safety at least equivalent to that of current generation LWRs.

Therefore, it is proposed that a set of event categories, corresponding to events which must used for design, siting and emergency planning purposes, be defined. Events to be included in each of these categories would be selected deterministically supplemented by insights gained from a PRA. The events selected could then be used as a basis for calculating source terms, evaluating the safety characteristics of the proposed designs and assessing the adequacy of their containment systems and offsite emergency planning. Each of the three DOE advanced designers has proposed such a set of accidents to be used in their designs. Their approach to accident selection is outlined in Enclosure 3.

The following are the staff proposed event categories and their associated description:

Event Category I (EC-I)

This category of events would be equivalent to the current Anticipated Operational Occurrences (AOOs) class of events considered in LWRs. The frequency range for these events goes down to approximately 10^{-2} per year, which corresponds to the frequency of events which may be expected to occur one or more times during the life of the plant. These events would be analyzed similar to what is done for LWRs to demonstrate compliance with Appendix I to 10 CFR 50 and 40 CFR 190.

Event Category II (EC-II)

This category of events for advanced reactors would be equivalent to the current DBA category for LWRs and would be selected consistent with the selection of an LWR DBA envelope. Specifically, events in EC-II would:

i) Be selected using traditional engineering judgement, complemented by PRA methods which would include internal events down to a frequency of approximately 10^{-4} per year (10^{-4}/yr is based upon ensuring that any event that is expected to occur over the lifetime of a population of reactors is included),

ii) Include a traditional selection of external events and

iii) Be subject to single failure criteria and other traditional conservatisms (no credit for non-safety grade equipment, etc.). Events within this category would require conservative analysis as presently done for LWRs.

Event Category III (EC-III)

This category of events corresponds to those severe events beyond the traditional DBA envelope which should be used by designers in establishing the design bases for these designs. The staff believes that the identification and use of such an event category is consistent with the Commission's Severe Accident Policy statement and is justified for advanced reactors, particularly those proposing the use of a mechanistic calculation of source terms and a shift in emphasis from accident mitigation to plant protection. The events in this category would be selected using engineering judgement, complemented by PRA. This is consistent with the guidance provided in the Commission's Safety Goal and Severe Accident Policies which encourage the use of PRA methods to supplement engineering judgement and deterministic (non-mechanistic) analyses.

Specifically, events in EC-III would:

i) Include internal events (less likely initiating events plus multiple failure events) down to a frequency of approximately 10^{-7} per year (10^{-7}/yr is based upon ensuring that the cumulative effect of events below 10^{-6}/yr are considered in assessing compliance with the Commission's proposed performance guideline of less than a 10^{-6}/yr frequency of a large release of radioactive material to the environment). External events beyond those included in EC-II would be included consistent with their application to future LWRs (currently being developed as part of implementation of the Commission's Severe Accident Policy).

ii) Include, using engineering judgment, additional bounding events to account for plant specific uncertainties (Enclosure 3 outlines an approach, based on identifying bounding plant states and challenges to safety functions, for deriving bounding events. In addition, a draft list of bounding events for the three DOE advanced reactor concepts is provided in Enclosure 3 as an example).

In selecting the events to be included in EC-III, the design would be specifically reviewed to identify those events with the potential of a large release, core melt, or reactivity excursion to ensure adequate prevention or protection is provided for these events before they could be excluded from this category. EC-III events could be analyzed on a best estimate basis.

Event Category IV (EC-IV)

This category of events is intended to be used in the assessment of the need for offsite emergency planning. It includes internal events of similar frequency to those events considered in the basis for the emergency planning zones and requirements

for LWRs (described in NUREG-0396 "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of LWRs"). These events would be analyzed in a PRA and would be used as described in Section II.B.4.

Event Frequencies

The staff recognizes that large uncertainties may exist in PRA results, especially in the lower frequency ranges. Therefore, in selecting and analyzing the events in the above categories, consideration must be given to the treatment of uncertainties. Accordingly, where the above event categories include in their definition a frequency value, this frequency value is intended to be a guideline only and is not to be considered a rigid limit for which compliance must be rigorously demonstrated.

Application to Modular Reactor Designs

In analyzing each event from the above event categories (in accident analysis and in assessing compliance with the criteria in Sections II.B.2, 3, and 4 of this paper), a determination must be made as to whether or not it applies to all reactor modules simultaneously or to one module only. In addition, in determining the events to be included in EC-I through the EC-IV and in assessing the risk from a plant (where a plant consists of more than one module), the probability of certain events occurring must be increased to account for the multiple modules.

Policy Considerations

The proposed criteria raise the following policy consideration:

- Does the Commission agree that the above defined event categories are necessary and sufficient to describe those events to be used for design purposes, including siting source term, containment and emergency planning considerations?

II.B.2) Siting Source Term Calculation and Use

Issue

Can the source terms used for siting be calculated using mechanistic analyses?

Staff Recommendation

The three DOE advanced reactor designers have proposed, to varying degrees, the calculation of a siting source term (that quantity and form of radioactive material available for release

to the environment after an accident) different than is currently done for LWRs. The MHTGR has proposed a mechanistic approach by analyzing the accidents to be considered in the design and calculating the release of radioactive material resulting from those accidents (in lieu of assuming a more arbitrary large release into containment as is done for LWRs using TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites"). The LMRs have proposed more arbitrary siting source terms to bound the release from accidents considered in the design; however, the magnitude of these source terms is much less than the TID-14844 LWR assumed source term. These different approaches all have as their goal being able to make siting determinations with source terms more in line with the characteristics of their designs. The staff believes source terms can be developed for advanced reactors based on mechanistic analysis, provided that (1) those source terms are used in conjunction with dose guidelines consistent with those applied to LWRs, (2) the events considered in the mechanistic analysis are selected to bound credible severe accidents and design dependent uncertainties and (3) the performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit mechanistic analysis. This will provide a more realistic estimate of source terms and give advanced reactor designers incentive to develop designs which minimize releases. The criteria proposed for application in the calculation and use of a mechanistic siting source term are listed below:

<u>Siting Source Term Calculation</u>

- Using the EC-II spectrum defined previously, perform a conservative evaluation of EC-II scenarios and calculate source.

- Using the EC-III spectrum defined previously, perform a best estimate evaluation of EC-III scenarios and calculate source.

- Ensure sufficient data exists (through R&D program and/or prototype testing) on reactor and fuel performance under EC-II and III conditions to provide adequate confidence in the mechanistic analysis methods used.

- Ensure none of the EC-II and EC-III scenarios are on threshold where a slight change in assumptions or uncertainty can cause an unacceptable change in source.

<u>Siting Source Term Use</u>

To allow the use of mechanistic analysis for siting source term selection, the following dose guidelines would apply for siting determination:

Dose Guidelines		Meteorology
EC-II	10% of 10CFR100	conservative
EC-III	10CFR100	conservative

The dose guideline specified above for EC-II is based upon maintaining an equivalent dose guideline as LWRs where they use mechanistically calculated source terms (i.e. where the LWR Standard Review Plan allows the use of mechanistically calculated source terms in analyzing accidents it specifies offsite dose must be a small fraction of 10CFR100 guidelines, which is generally interpreted as 10-25 percent of the 10CFR100 dose guidelines). The dose guideline specified above for EC-III is based upon applying the same siting dose guideline as is applied to LWRs (10CFR100) to those events which are being analyzed in place of the traditional non-mechanistic LWR source term (i.e. EC-III events are the severe events which in an LWR have traditionally been predicted to result in a core melt, and which for LWRs, lead to the establishment of the non-mechanistic TID-14844 source term). In analyzing releases from both EC-II and EC-III events, the use of suitably conservative meteorology is proposed so as to be consistent with traditional LWR siting calculations.

These proposed criteria on siting source term calculation and dose guidelines would be used in conjunction with the traditional assessment of site suitability using the guidelines of Reg. Guide 4.7 "General Site Suitability Criteria for Nuclear Power Stations" for factors such as population distribution, meteorology, etc. The criteria described in this paper are not intended to modify any of the other NRC siting guidelines as described in Reg. Guide 4.7.

Policy Consideration

The proposed criteria raise the following policy consideration:

- Does the Commission agree that source terms calculated on a mechanistic basis can be used as described above for siting determination?

II.B.3) Containment

Issue

What criteria would an advanced reactor design have to meet to receive design certification without a high pressure, large volume containment building like those found on LWRs?

Staff Recommendation:

The three DOE advanced reactor designers have proposed a containment system different from those found on LWRs. In the MHTGR, the fuel coatings provide the containment function in the plant. In the LMRs, a guard vessel that surrounds the reactor vessel and the reactor vessel closure head provide for containment of fission products and radioactive sodium (the SAFR design also has a secondary low pressure containment structure over the reactor head to contain leakage through mechanical seals in the head). Each designer has proposed a different way to demonstrate the fission product retention capability of his design. PRISM proposes to utilize testing of a prototype plant (preferably on a DOE site) for this demonstration, SAFR proposes to utilize testing of the first commercial unit (not necessarily on a DOE site) for demonstration and the MHTGR has proposed that R&D program results, combined with the normal startup testing of the first commercial unit (not necessarily on a DOE site), will be sufficient for demonstration.

The staff recognizes that a design without a containment building represents a significant departure from past practice on LWRs and that under certain situations LWR containment buildings have proven to be an effective component of the defense in depth approach. Therefore, designs which deviate from such practice need to be reviewed to ensure that an equivalent level of safety as current generation LWRs is maintained and that uncertainties in design and performance are properly accounted for. The staff believes that such designs are possible, although the ultimate acceptance of such designs will require extensive review, testing and demonstration. Accordingly, the staff proposes criteria to be met in order to certify a reactor design without a containment building with the understanding that in reviewing a design against these criteria, a large burden will rest with the applicant to demonstrate compliance, particularly in view of the uncertainties associated with a new design.

Specifically, the following are proposed criteria that advanced reactor designers must meet for NRC certification of a design without a containment building:

1. The design should contain multiple barriers to radiation release which limit radiation release for EC-I, EC-II and EC-III events, at least equivalent to that on current generation LWRs. Specifically:

- 10CFR50, Appendix I, and 40CFR190 limits should be met for normal operating conditions, including events in EC-I.
- The 10% of 10CFR100 and the 10CFR100 dose guidelines should be met for the EC-II and EC-III events, respectively, as described in Section II.B.2 above.

2. The fission product retention capability of the design must be demonstrated via a testing program utilizing a full size prototype plant (i.e. consisting of at least one reactor module and its associated systems, structures and components necessary to demonstrate safety). Such testing should be done at an isolated site (such as the National Reactor Testing Station) and the prototype plant should conform to the same regulations and standards as the design to be certified. The testing program should generate plant performance data sufficient to validate safety analysis analytical tools over an extensive range of operating and accident conditions considered in the design (EC-I, II and III), including an assessment of the response of the plant safety features over those conditions which may vary over the life of the plant (such as fuel burnup). However, this test program is not intended to be used to demonstrate component and system reliabilities nor exceedingly rare events that may have been accounted for in the design basis.

3. Different emphasis and types of QA, surveillance, in-service inspection and in-service testing over and above that traditionally employed on LWRs should be provided, as necessary, to ensure that the new and innovative systems, structures, and components which contribute to performing the containment function are, in fact, built, operated, and maintained over the life of the plant in a fashion commensurate with their safety function. (For example, the MHTGR fuel quality may require special attention due to its role in limiting the release of FP.)

4. Protection of safety related systems, structures, and components from sabotage and external events should be provided at least equivalent to that for current generation LWRs.

5. The design should take specific measures to ensure that no core melt accidents, accidents with significant positive reactivity feedback or other accidents with the potential of a large radiation release (such as graphite fires) are in the EC-I, EC-II or EC-III spectrum.

6. An assessment of the potential improvement in safety if a containment building were added would have to be made. Judgement would then used to determine the need for a containment building based upon the cost and change in risk.

The above criteria are intended to maintain at least the same level of protection of the public and environment (by specifying equivalent dose guidelines and protection) as is provided by current generation LWRs. In addition, for acceptance of a design without a containment building, these criteria would require demonstration (via a full size prototype test at an isolated site) of the fission product retention capability of the design. Requiring such demonstration testing is considered necessary to compensate for removal of the traditional (and testable) containment building. Such testing will help ensure that licensed plants of that design have adequate fission product retention. In fact, the potential of these advanced designs to prevent core damage over an extensive range of low probability events allows such integrated full scale testing to be done, whereas the testing of the response of containment buildings (for those designs which utilize containment buildings) to low probability events is usually limited to less than completely prototypic conditions. These criteria will allow designs that propose to withstand severe or bounding events without the need for a containment building (with due consideration for uncertainties) to be licensed and certified.

It should be noted that with respect to prototype testing, a second option (in addition to testing a prototype reactor at an isolated site) was also considered. This second option would allow the prototype reactor to be built and tested at a standard site (any site consistent with the requirements of Reg. Guide 4.7) provided that a containment building and traditional emergency planning were provided for the prototype. Although this option is technically viable and is intended to provide the industry with more flexibility in developing and locating a prototype, its application has several practical problems which potentially may make it more difficult. Major problems with this option are:

1) Reg. Guide 4.7 was intended to apply to sites for reactors which have been demonstrated. Specifically, it states that it is limited to LWRs and HTGRs. Therefore, this option could lead to a license application for a first of a kind reactor in a potentially moderate population area whose performance is sufficiently uncertain that special tests and a containment are needed to assure its safety. This could result in much opposition and protracted licensing hearings.

2) Questions regarding the containment's affect on the performance of the passive decay heat removal system would need to be resolved such that the testing would in fact confirm the design to be certified.

3) The basis for the containment design would have to be established. In addition, it is likely that the containment design would have some unique features, due to the fact that its design would have to allow demonstration of the passive decay heat removal system (as stated in 2 above). These unique features may require testing and qualification prior to their acceptance.

4) The characteristics of the site may limit the extent of and type of testing which could be performed. This could result in additional separate effects testing being required.

If this option is of real interest to a potential applicant and if the Commission wishes to have such an option available, the staff would need to develop additional guidance addressing factors important to its implementation, such as the containment design basis, containment isolation and qualification requirements and testing program objectives and strategy.

Policy Consideration

The proposed criteria raise the following policy consideration:

- Does the Commission agree that a design without a containment building could be certified, provided that the above stated requirements can be met without it?

II.B.4 Offsite Emergency Planning

Issue

Are there conditions under which preplanned offsite emergency evacuation and drills, as currently required for LWRs, need not be required by the NRC?

Staff Recommendation:

The three DOE advanced reactors have objectives of achieving very low probabilities (1.0×10^{-6} per year or less) of exceeding the Environmental Protection Agency (EPA) lower level Protective Action Guidelines (PAGs) of 1 Rem whole-body and 5 Rem thyroid at the site boundary. For these designs, DOE has proposed the plume exposure pathway emergency planning zone (EPZ) be encompassed by the plant exclusion area boundary and, therefore,

offsite emergency planning requirements can be greatly reduced. In the future, some reduction in the ingestion pathway EPZ may also be requested; however, at this time, only changes in the plume exposure pathway and its associated evacuation requirements are being addressed. In essence DOE proposes that these advanced reactors are so safe with their passive reactor shutdown and cooling systems and with core heatup times much longer than LWRs, that the EPZ radius can be reduced to the site boundary and that detailed planning and exercising of offsite response capabilities need not be required by NRC regulation. This does not mean there will be no offsite emergency plan developed, but rather that such a plan can have reduced detail concerning movement of people and need not contain provisions for early notification or exercise of the plan.

The staff believes that restricting the plume exposure pathway EPZ to the site boundary is equivalent to not requiring offsite emergency planning. Since the current policy of the Commission is that offsite emergency planning is a requirement for the licensing and operation of a nuclear power plant, the staff believes that the Commission should address the DOE proposals as a request for a change in policy rather than an adjustment of the EPZ size.

Currently, offsite protective actions are recommended when a situation occurs that could lead to offsite doses in excess of the PAGs, which are 1-5 Rem whole-body and 5-25 Rem thyroid. At the lower projected dose, protective actions should be considered. At the higher projected dose, protective actions are warranted (Dose that has already been accumulated prior to the decision on whether to take protective actions is not considered as part of this planning decision). In the past, the Commission has not required offsite emergency planning in those situations where the lower level PAGs were not expected to be exceeded. For example, emergency planning for research reactors is restricted to the area around the reactor where the lower level PAGs are expected to be exceeded. This is usually within the owner controlled area. For fuel cycle facilities, the proposed rule on emergency preparedness exempts those facilities where the lower level PAGs will not be reached outside the owner controlled areas. Therefore, there is a precedent for not requiring offsite emergency planning, beyond simple notifications, where warranted by operation. Response of certain offsite agencies into the owner controlled area (e.g. police, fire, medical) is traditionally considered part of the onsite planning.

The staff believes that emergency planning requirements for advanced reactors should be based upon the characteristics of those designs. This principle is similar to that in the emergency planning rule (10CFR50.47) which states that the size

of the EPZ for HTGRs can be determined on a case-by-case basis. In addition, the power level of each advanced reactor module is much smaller than a conventional LWR and, based on size alone, some reduction in the EPZ radius may be warranted similar to what has been done on the existing small size LWRs. In addition to these considerations, it is the staff's judgement that a plant's ability to prevent significant releases of radioactive material (particularly the prevention of core melt) and to provide long times prior to releases for all but the most remote probability events should also be reflected in any emergency planning requirements. Accordingly, the staff proposes criteria which consider such ability, consistent with evaluating a range of events similar to those evaluated for LWRs.

Specifically, the staff proposes the following criteria as a guideline for the advanced reactor designs in order for NRC to accept the DOE proposal of no traditional offsite emergency planning (other than simple notification):

> While an offsite emergency plan would still be required, such a plan would not have to include early notification, detailed evacuation planning and provisions for exercising the plan if:
>
> - the lower level PAGs are not predicted to be exceeded at the site boundary within the first 36 hours following any event in categories EC-I, II and III, and
>
> - a PRA for the plant, which includes at least all events in categories EC-I through EC-IV, indicates that the cumulative mean value frequency of exceeding the lower level PAGs at the site boundary within the first 36 hours does not exceed approximately 10^{-6}/year.

The above criteria give credit for designs which provide long times prior to significant radiation release. For designs such as these, the staff believes that because sufficient time is available, prompt notification of offsite authorities will permit effective evacuation without the level of preplanning currently required for LWRs. The bases for the two proposed criteria are discussed below.

The first criterion ensures that all events considered for design and siting purposes do not lead to offsite doses in excess of the PAGs early in the event sequence. Based on historical "ad hoc" evacuations in the United States (which have ranged between 2-8 hours), the staff believes that 24 hours is sufficient for local agencies to take protective actions (e.g. shelter or evacuation) and that in these cases preplanning does

not substantially reduce the risk to the public. The 24 hours, combined with 12 hours for the plant staff to diagnose the event and attempt corrective action prior to initiating evacuation/sheltering, is the basis for the 36 hour criteria.

The second criterion ensures that events beyond those considered for design and siting purposes (of a frequency similar to those events considered in NUREG-0396 for LWR emergency planning purposes) are considered for advanced reactor emergency planning purposes and that they do not contribute substantially to overall risk.

Policy Consideration

The proposed criteria raise the following policy consideration:

Does the Commission agree that emergency planning, modified as discussed above, is sufficient for plants that meet the above criteria?

Additional Considerations:

Impact of Commission Decision

In the advanced reactor designs under review, DOE has attempted to develop potentially safer designs using simple, passive safety systems and has emphasized core-melt prevention. The containment systems proposed by DOE are largely a result of this philosophy. The Commission decision on the issues of siting source term and containment, including defense in depth considerations such as the balance between prevention and mitigation, could impact the emergence of designs that aim for a high core-melt prevention capability, which in many cases may lead to mitigative systems different from those found in the current generation of LWRs.

In addition, the advanced reactor designs have as a goal not exceeding the EPA Protective Action Guidelines at the site boundary during accident conditions. As a result, they propose that traditional offsite emergency planning is not necessary for their designs. The Commission decision on this issue will indicate the degree to which new designs that have characteristics that affect the need for traditional offsite emergency planning can be given credit for these characteristics.

Application to Advanced Designs

The criteria proposed in this paper are of a preliminary nature for the purpose of providing licensing guidance to DOE early in the design process. Accordingly, they have not undergone the

formal public process associated with rules and other types of regulatory guidance. Therefore, the staff intends to provide, after completion of the pre-application reviews of the three DOE advanced concepts, a recommendation on the need to codify via rulemaking the general licensing approach and framework for advanced reactors discussed in this paper. This would then have the effect of formalizing the work done at the conceptual design review stage and contribute toward stabilizing the licensing requirements for reactor designs significantly different than current generation LWRs. In the interim, as the design of the advanced concepts proceeds, it is the staff's intention to assess the progressively more detailed design against the criteria proposed in this paper. If, at a later date, the designs were not able to meet the criteria, the designers would be expected to compensate in their design. Conversely, if at a later date additional information becomes available which would indicate that a change in any of the events included in event categories I thru IV is warranted, then a compensating change should be made. Staff review to date indicates that the DOE advanced concepts under review appear capable of achieving the proposed criteria. The staff's assessment in this regard will be documented in the SERs the staff is preparing on each design and draft versions of the SERs will be provided to the Commission over the next several months to assist in the review of the proposed criteria.

Peer Review

The staff conducted a peer review of the proposed approach and criteria discussed in this paper. The participants and conclusions of this peer review are summarized in Enclosure 4.

Coordination: The Office of the General Counsel has reviewed the contents of this paper and has no legal objection but strongly believes that many of the important issues addressed in this paper should be resolved by rulemaking well in advance of any design certification rulemaking. ACRS has been briefed on the contents of this paper and is expected to issue a letter on this subject in the near future.

Recommendation: The Commission:

(1) endorse the proposed approach and criteria in this paper for staff use in preparing guidance to DOE on the key licensing issues associated with their advanced reactor designs under review.

(2) note that the staff intends to apply its proposed criteria only to the DOE sponsored advanced reactors.

(3) note that the staff expects its proposed criteria to result in a level of safety for the DOE reactors at least equivalent to the current generation of LWRs and well within the safety goals.

Victor Stello, Jr. Executive Director
for Operations

Enclosures:
1. Proposed Designs Submitted for Review
2. Components of Defense in Depth
3. Bounding Plant States and Failures for the Advanced Reactors for Determining EC-III Events
4. Summary of Peer Review of Commission Paper on Key Issues Associated with Advanced Reactors

Commissioners' comments or consent should be provided directly to the Office of the Secretary by c.o.b. Wednesday, August 31, 1988.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Friday, August 5, 1988, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

DISTRIBUTION:
Commissioners
OGC
OI
OIA
GPA
REGIONAL OFFICES
EDO
ACRS
ASLBP
ASLAP
SECY

ENCLOSURE 1

PROPOSED DESIGNS SUBMITTED FOR REVIEW

Modular High Temperature Gas-Cooled Reactor (MHTGR)

The MHTGR design uses a 350 Mwt standard reactor module designed by GA Technologies. The MHTGR Plant design using the standard reactor module is being developed in conjunction with Gas Cooled Reactor Associates, Stone & Webster and Bechtel.

The reactor module uses a steel reactor vessel and an external steam generator located below grade in concrete silos. The reactor building, consisting of these silos and a building above grade, does not provide a fission product containment function during accident conditions.

The fuel design is similar to Fort St. Vrain. The MHTGR fuel is designed to retain fission products up to 1600 C - 1800 C. DOE contends that the fuel, through its high temperature performance capability, provides the major containment function in the plant for design basis and beyond design basis events. In addition, DOE contends that the MHTGR can withstand a wide range of low probability events with little or no fission product release, including station blackout, ATWS events, full withdrawal of all control rods, and multiple operator errors. Thermal response of the plant under core heatup conditions is slow.

Key features of the design are passive reactor shutdown characteristics (i.e. large negative temperature coefficients) and a passive decay heat removal system that utilizes natural draft air flow from the atmosphere to the reactor cavity to cool the reactor vessel and thereby remove decay heat. In addition, DOE believes that even if this system is lost, adequate decay heat removal can be achieved by radiation and conduction to the reactor cavity concrete and surrounding earth.

DOE contends that releases of radioactive material during accident conditions are low enough that no conventional containment building or offsite evacuation plans/drills are needed. DOE also contends that the source term(s) utilized in the siting determination should be selected on a mechanistic basis to give credit for the enhanced safety characteristics of the MHTGR.

Power Reactor Inherently Safety Module (PRISM)

The PRISM plant design uses a 425 Mwt standard reactor module designed by General Electric (GE). The PRISM plant design using the standard reactor module is being developed by GE with Bechtel as the supporting architect engineer.

Sodium is the primary coolant with a pool type primary system (all primary system components are located within the reactor vessel). The reactor module is located below grade. The fuel is U/Pu/Zr metal similar to that used in DOE's EBR-11 reactor.

The containment is a hermetically sealed low pressure, low volume, leak-tight barrier that employs a factory fabricated containment vessel around the reactor vessel, reactor head closure assembly and IHX tubes as the low leakage barrier

against postulated release of the core fission products. In addition, the containment vessel is located close to the reactor vessel to assure that primary coolant leaks from the reactor vessel do not result in loss of core cooling (i.e., the containment vessel functions as a coolant retaining guard vessel).

As for the MHTGR, key features of this design are passive reactor shutdown characteristics (via large negative temperature coefficients) and a passive decay heat removal system with natural draft air flow from the atmosphere for cooling the reactor vessel.

DOE contends that the plant is designed to withstand a wide range of low probability events (e.g., station blackout, transient overpower w/o scram, loss of flow w/o scram, etc.) with little or no radiation releases. Due to the large thermal capacity of the sodium pool, the heatup of the core is slow, when compared to LWRs, even for cases where no decay heat removal system is assumed operating.

The public risk, due to off-site releases from the containment associated with postulated accidents, is claimed by DOE to be low enough that there should be no need for conventional off-site evacuation plans and drills.

Sodium Advanced Fast Reactor (SAFR)

The SAFR design utilizes a 900 Mwt standard reactor module designed by Rockwell International (RI). The SAFR plant design using the standard reactor module is being developed by RI with Becthel as the supporting architect engineer.

Sodium is the primary coolant, with a pool type primary system (all primary system components are located within the reactor vessel). The reactor vessel is located above grade. The fuel is U/Pu/Zr metal similar to that used in DOE's EBR-II reactor.

The containment is a low pressure, low volume barrier which utilizes the reactor guard vessel and reactor vessel closure head. The reactor guard vessel also functions to assure that primary coolant leaks from the reactor vessel do not result in loss of core cooling. A secondary containment building also functions as a barrier to the release of fission products through mechanical seals in the reactor vessel closure head.

As for the MHTGR and PRISM plants, key features of the SAFR plant are passive reactor shutdown characteristics and a passive decay heat removal system that uses natural draft air flow from the atmosphere to cool the reactor vessel.

DOE contends that the SAFR plant is also designed to withstand a wide range of low probability events (station blackout, ATWS, etc.) and has thermal characteristics such that the heatup of the system is slow due to the large inventory of sodium.

Radioactive releases during accident conditions are low claimed by DOE to be low enough that conventional offsite evacuation plans/drills are not needed.

ENCLOSURE 2

COMPONENTS OF DEFENSE IN DEPTH

COMPONENTS OF DEFENSE IN DEPTH

PREVENTION

- Reliable plant systems:
  - ° reduce challenges to safety systems
- Reduce potential for human error*
- Conservative design:
  - ° Plant performance
  - ° Plant performance
  - ° Barrier integrity
- Control stability
- Quality assurance:
  - ° Design
  - ° Construction
  - ° Operation
- Good Oper./Maint. and training
- Safeguards and Security
- Supporting R&D and testing

PROTECTION

- Reliable independent redundant safety systems:
  - ° Reactor shutdown (active**/passive*)
  - ° Decay heat removal (active**/passive*)
- Maintain integrity of barriers to release of radioactive material under:
  - ° EC-I
  - ° EC-II
  - ° EC-III*
  - ° EC-IV*
  - ° Enhanced fuel integrity*(MHTGR)
  - ° Double reactor vessel*(LMRs)
- Long response time*
- Control stability
- Minimize need for human intervention*
- Emergency procedures

MITIGATION

- ESFS**
  - ° Spray systems
  - ° Filtering systems
  - ° Cooling systems
- Conventional containment building**
- Physical phenonema
  - ° FP holdup
  - ° FP plateout
  - ° FP decay
- Long response time*
- Emergency Procedures

EMERGENCY PLANNING

- Preplanned Evacuation/Shelterin
- Ad hoc evacuation* (Long response time)

* Key features in defense in depth that are utilized to a greater degree in advanced reactors than in current generation LWR designs.

** Key features in defense in depth that are utilized to a lesser degree in advanced reactors than in current generation LWR designs.

ENCLOSURE 3

Bounding Plant States and Failures for the Advanced Reactors

For Determining EC-III Events

Bounding Plant States and Failures for the Advanced Reactors for Determining EC-III Events

Introduction

The advanced reactor designers have proposed selection of accident categories and initiating events as described below. The staff believes that the selection of accidents, particularly for beyond current design basis events, cannot be made solely with PRA methods due to the large uncertainties.

Specifically, the following appear to be the major sources of uncertainty in the conceptual designs:

1) The limited performance and reliability data for the critical systems, mainly the passive decay heat removal system using atmospheric air to cool the reactor vessel. This is a new concept and is utilized in all three designs. Other key systems that need additional demonstration are the high temperature performance of the fuel in the MHTGR and the inherent negative reactivity feedback mechanisms in the Liquid Metal Reactors (LMRs);

2) The lack of a final design which limits identification of initiating events and dominating sequences;

3) Unverified analytical tools used to predict plant response;

4) Incomplete industry codes and standards for the unique aspects of the designs;

5) The state of the art of supporting technology relevant to the new designs;

6) Extrapolation of R&D results to a full size unit; and

7) Significantly less design, construction and operating experience as compared to LWRs;

In selecting the bounding states and failures, provision must be included which provides for, on a plant specific basis, a sufficiently conservative test of the design to account for plant specific uncertainties. Accordingly, the set of bounding events selected for consideration at the conceptual design stage should provide for a sufficient test of the conceptual designs such that accurately knowing the failure modes and failure probabilities of the safety features of these designs is not critical to assessing or understanding their safety. This will then provide confidence that the use of mechanistic source terms bounds expected releases and that the proposed containment system is conservatively tested. This will also allow an assessment of the licensability given their conceptual nature and the many uncompleted supporting R & D programs. An approach for selecting such events has been proposed by each of the designers and by the staff as described below.

Designer Proposals

Advanced reactor designers have proposed a set of events for application to their designs for the purpose of siting source term selection and containment and emergency planning evaluation. The designers have proposed these events be selected as follows:

- MHTGR

  Using PRA results at the conceptual design stage, accident sequences that have a probability down to $5x10^{-7}$/yr are considered. Events in this probability range are selected by evaluating accident sequences down to 10^{-8}/yr. Uncertainties in their probability are estimated and if the estimated uncertainty causes the event to frequency to exceed $5x10^{-7}$/yr then the event is considered.

- LMRS

  Traditional DBAs are evaluated similar to what was done on the Clinch River Breeder Reactor (CRBR). For accidents beyond DBAs, station blackout, and ATWS events under transient overpower (rod withdrawal), loss of forced circulation and loss of normal heat sink conditions are considered. These were the severe events considered for the CRBR.

Staff Recommendation

The staff recommends a selection of bounding plant states and failures consistent with, although more conservative than, that proposed by the designers. This approach complies with guidance contained in the Commission's Safety Goal Policy Statement and conservatively accounts for uncertainties in event probability and equipment performance. This approach utilizes engineering judgement supplemented by PRA to account for uncertainties as follows:

- Use criteria defined for EC-I thru III.

- For EC-III bounding events, use engineering judgement to deterministically impose a set of internal plant states and failures which bound uncertainties in event frequency and failures modes. These additional states and failures would be selected to bound reactivity insertion, heat removal, loss of primary system integrity, chemical/reaction (Sodium/Water reaction or air/graphite-water/graphite reaction), supporting systems failure and loss of coolant flow/inventory events by:

  - selecting bounding plant states (specified by system pressure, temperature, flowrate, etc.) during the challenges to the safety functions,

  - assuming non-safety grade equipment fails (either as an initiator or in response to the initiating event) in a way that exacerbates the accident to the maximum degree physically possible, unless a lesser degree can be justified. This will account for any uncertainties due

to using commercial grade procurement and construction and the lack of NRC inspection and technical specifications on this equipment.

- assuming failure of safety grade equipment for a period of time consistent with previous experience unless a lesser time can be justified (bounds uncertainties in failure probabilities of safety grade equipment)

- assuming multiple human errors consistent with events that have actually occurred

- allowing a reasonable time to recover from initiating events where no plant damage has occurred (ATWS, station blackout, loss of all cooling).

Example of Application of Staff Proposed Bounding Event Criteria

The following plant states and failures are presented for each of the three advanced designs as examples and are intended to deterministically bound the uncertainties in plant failure modes and frequencies such that these these uncertainties are not a decisive factor in the assessment of overall plant safety. This will then allow a conservative assessment of source terms and the adequacy of containment and offsite emergency plans at the conceptual design stage. A final list of these events for each advanced concept will be provided in the SERs. These plant states and failures provide bounding conditions and challenges to the following plant safety functions essential to prevent fuel failure:

- Control reactivity
- Remove decay heat
- Control chemical reactions
- Maintain plant support systems
- Maintain plant integrity
- Maintain adequate coolant inventory/flow

MHTGR

For the MHTGR, the plant response to the following challenges to the safety functions must be examined with the plant in the following states (as applicable):

- Only inherent reactivity feedback (assume failure of scram system);

- The primary coolant system in a pressurized and depressurized state (i.e. integrity may not be maintained); and

- All primary coolant flow conditions (i.e. with or w/o forced circulation)

Safety Function	Bounding Challenge
Reactivity	Inadvertent withdrawal of all control rods for "x" hrs. (one module)

Decay Heat Removal	Loss of all DHR systems for "x" hrs. (one module)
Chemical Attack	Rupture of 25% of SG tubes with failure to isolate or dump SG. (one module)
Plant Support Systems	Station blackout for "x" hrs. (all modules)
Plant Integrity	Double ended guillotine break of cross duct. (one module)

The plant response to external events consistent with those imposed on LWRs should be examined.

LMR

For the LMRs, the plant response to the following challenges to the safety functions must be examined with the plant in the following states (as applicable):

- Only inherent reactivity feedback (assume failure of scram system), and

- All primary and secondary coolant flow conditions (i.e. with or w/o forced circulation).

Safety Function	Bounding Challenge
Reactivity	Inadvertent withdrawal of all control rods for "x" hrs. (one module)
Decay Heat Removal	Loss of all DHR systems for "x" hrs. (one module)
Sodium/Water Reaction	Rupture of 25% of SG tubes with failure to isolate or dump SG. (one module)
Plant Support Systems	Station blackout for "x" hrs. (all modules)
Plant Integrity	- Double ended guillotine break of IHTS pipe. (one module) - DEG break of primary pipe (one module) - Reactor vessel leak (one module)
Primary Coolant Flow	- Seizure of one primary pump (SAFR) (one module) - Instantaneous stoppage of power to one primary pump (PRISM) (one module)

The plant response to external events consistent with those imposed on LWRs should be examined.

ENCLOSURE 4

SUMMARY OF PEER REVIEW OF COMMISSION PAPER ON
KEY ISSUES ASSOCIATED WITH
ADVANCED REACTORS

A Peer Review Team was organized to obtain comments on the staff's proposal for addressing key licensing issues associated with the advanced reactor designs proposed by DOE. The following members participated in the peer review effort:

1. Joseph M. Hendrie, Brookhaven National Laboratory

2. Roger J. Mattson, Scientech, Inc.

3. Robert J. Budnitz, Future Resources Associated, Inc.

To focus attention on the fundamental issues, the following questions were posed to the members of the Peer Review Team.

<u>Fundamental Issues/Questions</u>

1. On what basis can events or event sequences be excluded from consideration in decisions regarding source term, containment, and emergency planning? What are the limiting events which need to be considered for the DOE advanced reactor concepts?

2. On what basis can a siting source term be different from that source term called for in 10CFR100 (TID - 14844 source term)?

3. On what basis can a conventional containment building be excluded from a reactor design?

4. On what basis can the requirement for preplanned offsite emergency evacuation be excluded?

5. How should "defense in depth" be applied on reactors with respect to prevention of severe accidents, containment design, and emergency planning?

6. On what basis can portions of 10CFR and LWR-SRP be considered not applicable to advanced reactors?

The discussions with the Peer Review Team took place during November and December 1987. Many comments received from them have been incorporated into the staff's proposed criteria. Each member of the peer review team documented his comments in a letter.

The attached table summarizes the comments from the peer reviewers to the questions posed to them. It should be cautioned that the table attempts to briefly summarize each of the peer reviewers comments; however, for completeness, the actual letters should be reviewed.

ISSUE	STAFF PROPOSAL	RESPONSE FROM PEER REVIEW TEAM		
		Hendrie	Mattson	Budnitz
1. Method for event selection/exclusion of certain events	Exclude events beyond EC-III from source term and containment system design basis.	Concurs with staff's proposal. Advises evaluation of consequences of extreme events.	Staff's approach is suitable, need to distinguish between internal and external events. Should ensure events unique to new designs are evaluated	Generally concurs with staff proposal.
2. Source term selection	Mechanistic analysis of events	Staff's proposal conservative and more rational than TID-14844	Staff's proposal is appropriate and incorporates LWR operating experience.	Staff's proposal pretty close to mark. Advanced reactor designs should not have the same source term.
3. Conventional containment building	Other containment systems acceptable if certain criteria can be met.	Staff's criteria are sufficiently rigorous for acceptance.	A conventional containment building is necessary to avoid problems related to public perception of safety.	Should not be a priori required for any power reactor. For good public policy, mitigative design features are needed.
4. Offsite Emergency Planning	Detailed preplanned evacuation not required if certain criteria can be met.	Could rely on "ad hoc" evacuation if sufficient time is available.	Bad public policy to remove preplanned offsite emergency evacuation.	Concurs with staff proposal.
5. Application of defense-in-depth	Maintain defense-in-depth principle with some shift between elements allowable.	Overall effectiveness of principle maintained.	NRC should emphasize defense-in-depth. Levels of defense-in-depth need not all be independent of one another to be effective.	Agrees with staff approach, advises more flexibility be provided to designers.
6. Exclusion of portions of 10 CFR and LWR-SRP	Promulgate new regulations for advanced reactors.	No particular problem.	Exclusion should be based on technical grounds.	No response.