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NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON ADVANCED REACTORS

Nuclear Regulatory Commission
Room 1046
1717 H Street, N.W.
Washington, D. C.

Thursday, January 30, 1986

The meeting of the subcommittee convened at 8:35 a.m., Dr. Max W. Carbon presiding.

ACRS MEMBERS PRESENT:

DR. MAX W. CARBON
DR. CARSON MARK
DR. CHESTER P. SIESS

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UNITED STATES NUCLEAR REGULATORY COMMISSIONERS'

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

THURSDAY, JANUARY 30, 1986

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P R O C E E D I N G S

DR. CARBON: The meeting will now come to order.

This is a meeting of the Advisory Committee on Reactor Safeguards, Subcommittee on Advanced Reactors.

My name is Carbon. The other ACRS members here today are Dr. Mark and Chet Siess, who should be here momentarily.

The purpose of the meeting is to review the HTGR design that was submitted to NRR by DOE and to determine the basis for selection of the design basis events.

Dr. El-Zeftway is the cognizant ACRS Staff member for the meeting.

Rules for participation in today's meetings have been announced in the Federal Register.

A transcript is being kept. It's requested that each speaker identify himself or herself, use the mike, and so on.

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

I have just a couple of comments to make before we begin.

This meeting is a consequence of a letter from Carl Neal to Ray Fraley on November 26, in which NRR and DOE asked to interact with the Advance Reactor Subcommittee on

HTGRs and LMR designs. At the present time, at least the ACRS is planning to go ahead with this. We are suffering like everyone else from some budget cuts, and we're going to have to make some decisions as to what we do and what we don't do. But for the present, at least, plans are to continue the interactions with your two groups. It will tentatively be our intent to pass on comments, obviously, during the discussions on days like this. We may very well forward written comments after discussion with the full committee, but this is not determined at the present time.

We have today's meetings scheduled on HTGR. Then there is a meeting scheduled for late February on the LMR.

One other comment. We're planning to charge right through to 2:00 o'clock and go without lunch. If this gives anyone any problems, let us know. It will be helpful on plan reservations for us, and so on.

Carson, do you have any comments?

DR. MARK: Not really. But generally, perhaps, the proposals for advanced reactors, of which this is one, and an attractive looking, in what degree does our participation affect the time scale? The time scale is really affected by when somebody gets down to business and says we want one. At that point, we need to come in, indeed. Right now we're interested, indeed, in hearing about the design thoughts which people have, what is it we

can do about advancing or promoting or accelerating or simplifying the process towards such a thing happening.

That's the kind of question I have in mind.

DR. CARBON: I think I can provide a partial answer at least.

We've encouraged NRC and NRR to work with DOE, early in the stage of design and so on. I know I personally am a strong believer that safety ought to come in early and be built in as part of the overall design aspects right from day one. I think it ends in a much better product, both from the standpoint of the economic side, the operational side, but also from the safety side, and I think this is part of the process of not only NRR interacting with DOE, but also DOE and NRR asking us for our comments.

If we were to have strong objections to what they were doing and these didn't show up till way late in the design stage, as a bare minimum, it would certainly lead to some bad inefficiencies and repetitious design effort and that sort of thing.

DR. MARK: Well, I'm going along on that hypothesis too.

DR. CARBON: Chet?

DR. SIESS: Max, I guess I've been looking at this in the context of what we've done in the past, reviewing a concept. Remember, we reviewed the concept of

upper head injection, ice condensors, things of that sort and picked up on certain things.

If you look at the conceptual design for the small HTGR, I see two issues that the ACRS could easily be involved in at some point which are crucial to the concept, and indeed, could be addressed.

Now there may be others. One of them is the question of whether there's a containment or not. This fits into the philosophy of what doses are, how do you accept extreme accidents and so forth.

I mention that, because that was an issue of the only other gas-cooled that we've reviewed to completion -- Fort St. Vrain.

When we say other gas-cooled, they had containments. There was a strong recommendation in the ACRS letter that the next one, a bigger one, ought to have a containment, and I think whether the design has a containment or not is a major point, because that affects the economic viability of it, and it's going to affect some of the conceptual approach.

I think another aspect of the concept that the ACRS ought to look at and agree with somebody on is the separation of a safety-related system from a nonsafety-related system. I think that accepting the idea that something of this sort was not safety-related, and this

one is, again is important to the economic viability and safety viability of the design.

I'd hate to try to go through the definitions of "safety-related" or "important to safety," which I'm not sure are settled, but there are things that we didn't hinbk were safety-related on lightwater reactors at one time, that we not think are safety-related, the outstanding example being aux feedwater. And now we're looking at all sorts of things that could be an initiator of a transient, an initiator of an accident, or a challenge that isn't safety-related that could be a challenge. I think of safety-related as only those things that are required to mitigate an accident and so forth. But again, that separation, I think, is conceptual.

I think it's important to the design, the viability of the design. And I think it's the sort of thing that the ACRS would have an opinion on, and they're not going to have a good opinion on it, until they understand a lot about the design.

So those are two things. There was a third one I had in mind. I guess the third point isn't all that clear, from what I heard. It may get cleared up today. We're still having a problem like water reactors in separating design basis accidents from severe accidents. The severe accident is being treated on a probabilistic basis. Design

basis accidents being a deterministic type of thing. And except for the severe accident policy statement, we don't have very much on severe accidents. I guess you could say, even with the severe accident policy statement, we don't have very much, because nobody knows how to relate it to a safety goal, because we don't have a safety goal.

So those are the kinds of things I think the ACRS could address, although the ACRS can probably never comment itself to what a future ACRS might do.

You get my point. A review of the concept and trying to find those features of the concept that are important, not the details of it, not whether it's a concrete vessel, a steel vessel or whatver.

DR. MARK: Chet, I think you put things beautifully. Just right. That's where we are and where we need to concern ourselves.

DR. SIESS: You know, there are going to be things like the purely passive concept, the walk away, the extremely low doses, and I guess if everybody could be convinced that the consequences are going to be small, I don't care what you do, you can shift gears and get into a different domain of thinking.

Now whether anybody can get to that stage or the conceptual review, I don't know, but anything that will keep moving in that direction would help.

DR. CARBON: Fine. I too share the view that you've put it very, very well, and also, share the technical views.

Let's go ahead then and call on Tom King.

(Slide.)

MR. KING: My name is Tom King, with the Office of Nuclear Reactor Regulation. I'm a Section Leader in the Safety Program Evaluation Branch in the Division of Safety Review and Oversight. My section has the responsibility for the advanced reactor reviews and interactions with DOE on both thge HTGR and the LMR designs.

I think Dr. Carbon summed it up well this morning. The purpose of these interactions is to work early on in the design process to factor in safety and licensing considerations into the design, so that we don't get further downstream and have to work these issues out and cause a rework of the design or patchwork-type fixes. We're trying to incorporate safety into the design right from day one.

These interactions are consistent with the Advanced Reactor Policy Statement which not only calls for the Staff to interact early but also request ACRS involvement in these interactions.

The purpose of today's meeting primarily is to acquaint ACRS Subcommittee members with the HTGR design. DOE and its contractors will be doing most of the talking

this morning. I'm just going to give a quick overview of where our review plans stand at the present time.

What we've agreed to with DOE is to review the conceptual design of the HTGR. This is a process that will take place approximately over the next two years, with the primary purpose being to develop and agree on criteria for the design, to assess the potential of that design for satisfying these criteria, to assess the R&D programs supporting that design. And at the end of the two-year process, to issue an SER and what we call a licensability statement.

At the end of this two-year period, we're planning, or we hope to be in a position then, to identify additional steps. Research, for example, that need to be taken for the NRC to be ready to process and HTGR application for either standard plant review or an actual commercial plant.

DR. MARK: What authority -- I'm not sure that's the right word. What authority, let me say it again, do you have -- supposing you come to the conclusion, which is a possible conclusion, that an HTGR, to be specific, need or need not have high level containment. What authority do you have then to represent NRR at the agency on that aspect of things?

MR. KING: When we would issue an SER, it would

be NRR's position on what we think of the design, which features we don't like or do like. We would certainly factor in ACRS comments, thoughts, positions on that, but it would not have the force of law like an actual license would, of DOE or the contractors could.

DR. MARK: It would still be subject to potentially endless debate, whether this was the correct view or not.

MR. KING: Potentially, it could. It could go on, but the idea is to try to get a consensus early on, a staff consensus and ACRS consensus and a designer consensus.

DR. MARK: That's certainly among the questions we're going to have to tangle with, I would suggest before very long, perhaps even today.

DR. CARBON: A couple of questions. One that sort of follows Carson's. How much interaction will you have, does your group have with the rest of NRR? When we end up here, you're in effect saying, well, tentatively NRR feels so-and-so about this. But you say will depend to a considerable part on whether it's just a small group of people, or whether a large part of NRR has had some sort of look or interaction here.

MR. KING: In the current organization, the new organization at NRR, we will be drawing upon a lot of people

in Dr. Spiess' division, the Safety and Oversight Division.

DR. CARBON: You are in that?

MR. KING: I am in that Division. My particular branch will have people involved. We have approximately 10 or 12 other people outside of my branch that will be involved. We're going to use NMSS for safeguards and security review. We will probably go to other people in NRR where they have expertise in specific areas like the fire protection expert is not in. If he's in another division, we will try to use him, but primarily, the review will be done in the SRO. They're assigned that responsibility in the new organization.

DR. CARBON: But is it a fair statement to believe that a good-sized segment of NRR will have been exposed to the thoughts and concepts in this and will have given at least a little preliminary thought to it.

MR. KING: A fair-sized segment of DSRO will have been exposed to it. The other divisions are vendor divisions, and there will be a limited number of individuals in those divisions who will have been exposed to it.

DR. CARBON: Will Spiess have reviewed it?

MR. KING: Yes.

DR. CARBON: How much involvement will Denton have in this? How much review will he do?

MR. KING: I can't speak for Mr. Denton, but our

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intention is to issue certainly things like the licensability letter over his signature.

In fact, as we go along, I think a lot of the Staff work and responses to the DOE on the various key issues that we're going to be looking at, our plan is to send those out under Dr. Spiess' signature. If there's a particular key one, for example, whether there is a containment or not a containment, I think that kind of issue may get elevated to Denton's level. But we'll have to see what the issue is and make that decision based on that.

We don't have a predetermined plan at this point as to who is going to sign out what, but it's going to be the Spiess-Denton level, depending on what the issue is. Our correspondence deals with that.

DR. CARBON: The second question refers to the last bullet there on the SER and licensability statement. Could you pin down what that licensability statement will do or say or mean, a little bit more?

MR. KING: Our intent at this point is for it to be a summary of what the major features are of that design, the major characteristics of that design we'd like to see to make it licensable. In other words, it will concentrate on those area that maybe we've had a problem with, or we'd like to see some change in.

Maybe the best way to look at it is a summary of what comes out of the SER, look at the whole design throughout.

There will be things that we agree with. There will be things we have problems with.

The licensibility statement in the division will be sort of a summary. Here are the major things that we would like to see changed to make this design licensable.

DR. CARBON: Will the thrust of it be to say, for example, it is our opinion that this is on a permanent basis and it could probably readily move forward and expect to receive a license somewhere down the road and we only have problems with two or three particular things and we should change those, or is it perhaps to be more simply a statement: the only suggestions we can make are these.

What is the SER going to say? It can't be final?

MR. KING: No. We are dealing with a conceptual design. It is not going to have a lot of detail in many areas.

We are not going to get into, for example, details of analysis codes and validation of those codes. But I think in a broad sense we are going to look at things like what codes we are going to say for an actual application. You are going to have to demonstrate how you

validate this code.

If we see areas in the R&D program where they need additional fuels data to support their fuels, materials data to support their reactor vessel, we are going to try to identify those, and the licensibility letter is going to summarize those and say here are the key areas where we feel additional work needs to be done; some changes need to be made to make this design licensable in the future for an actual application.

We are trying to give some early indication of where we see some weakness and see some additional information required.

DR. SIESS: Is it your intention at some stage between now and licensing to require a PRA? If so, at what stage? If not, why not?

MR. KING: The PRA is already planned. You will see it on the next viewgraph at the conceptual stage. It is scheduled to be submitted to us in September of '86.

DR. SIESS: That is just a schedule.

How helpful would a PRA be in addressing the two issues I have raised -- containment and separation of safety-related and nonsafety-related?

MR. KING: I think for the latter it would be very useful, identifying what are the initiators, the risks, the probabilities of those occurring, to see how much

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contribution to risk the balance of plant or the nonsafety grade portions of the plant really contribute.

DR. SIESS: Why not the first one?

MR. KING: Possibly for the first one, also.

DR. SIESS: I mean, the PRA was conceived to address somewhere a worst case accident, no matter how low the probability, and if the consequences of a worst case accident at any probability were sufficiently low, you would have a pretty good argument on the containment issue.

MR. KING: Yes, you would. In fact, if you do, say, a cost-benefit type analysis of adding containment versus not adding a containment, you are going to need PRA type work to evaluate the consequences, to do that kind of analysis.

So on both points it would almost be essential to have a PRA.

DR. CARBON: Move on.

(Slide.)

MR. KING: This is just a quick table that summarizes the major steps over the next two years in the interactions that we are planning to have. I won't go through everything, just point out a couple of things.

We have already had a number of briefings dealing with the licensing approach, top level criteria, selection of licensing basis, defense design basis events, that kind

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of thing.

The next step, beginning next month, is to interact with what we see as the major issues, decay heat removal and containment, those types of things.

I have got a column here for the NRC expected action date, where we give some feedback to DOE and its contractors.

And here in the last column on the right are the interactions we had planned with the ACRS subcommittee -- today's meeting covering these. We have another session on the HTGR scheduled in September to talk about the containment confinement and what we will call the balance of plant classification. That is the safety grade, nonsafety grade.

DR. SIESS: Those are the two items I mentioned, but let me comment.

You have an ACRS briefing, and you say the interaction with the ACRS subcommittee. That is a little different than what I said. I don't think interaction with the subcommittee is going to get you anywhere except in defining issues.

I think there should be an interaction with ACRS, and I think what you probably want -- and if you don't want it, you probably need it -- is an ACRS letter on the conceptual design.

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MR. KING: Yes, we had planned for that.

DR. SIESS: That is not something that is going to be accomplished simply by meeting with the subcommittee. I would guess it is not going to be accomplished with one meeting with the ACRS.

And so --

MR. KING: Let's talk about that a minute.

DR. SIESS: That schedule has got to be looked at.

(Slide.)

MR. KING: This is the last part of that same table. This covers FY '87 primarily, where we received the PRA in September, we received the preliminary safety information document, and the thrust in '87 is to review the PRA and the PSID. And out for ACRS we have TBDs.

One of the things we wanted to talk about today -- two things. One is are there any additional interactions or briefings you would like to have in FY '86 on any particular aspect of design?

The second is when should we get the full committee involved?

My own thought was we would get them involved when we get the PSID, which will describe the design, and then have one or two or whatever number of sessions it takes with the full committee because we would like a letter from

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them before we issue our SER on the PSID, which will be later on in FY '87.

I agree with you, but we haven't worked out the timing and the number of meetings yet.

DR. SIESS: I am trying to think to what extent ACRS questions or comments might affect what is done in completing the development of the reference concept, in doing the PRA or the PSID. I am not sure you want to wait until you have got the PSID and come in and get a lot of questions that could have been addressed earlier.

It seems to me you want to get some kind of a round with the full committee that has been brought up to some level of understanding and see what kind of concerns they have, then go back and see whether those concerns are being addressed or could be addressed in the PRA and the PSID. That would expedite things.

That isn't as simple as it sounds because sometimes it takes a document to get the ACRS attention. By that, I mean both a document from the applicant and from the staff.

MR. KING: I guess what I would like to suggest is maybe we could come back to this subject after you have heard the presentations on the design and the licensing approach and criteria. Maybe at the end of the day we could talk about what additional interactions there should be.

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I would appreciate your thoughts on what would be useful to the subcommittee and the full committee.

DR. SIESS: I agree.

(Slide.)

MR. KING: The last viewgraph is a quick summary of the review support the staff is using to conduct these reviews over the next two years.

Right now we have a small contract with MIT. That will be the fuel design and performance of the HTGR fuel. We have a contract with Oak Ridge and Brookhaven to perform independent analysis and assist us in the review of the submittals from DOE.

Research, although they don't have any money in FY '86, they have a lot of money carrying over from FY '85, and they are completing work on an HTGR handbook, which is primarily to document the state of the art in the HTGR materials, fuels, designs, foreign designs, just trying to get a good summary of the HTGR state of the art.

And then we are trying to maintain cognizance of the foreign activities. Oak Ridge has contacts with the German people involved in AVR and THTR, international conferences -- Pete Williams went to Germany in September for the International HTGR Conference -- cognizance of what is happening with the THTR startup and the German HTR-500 design, the large gas reactor activities going on at this

point.

DR. CARBON: Which country seems serious on HTGR activities at the present -- Germany?

MR. KING: The Germans.

DR. CARBON: Anyone else?

MR. KING: Japan is working on an HTGR, but its timeframe I believe is the late 1990s. It is a small one. I couldn't say more than that at this point.

With that I am going to turn it over to DOE, and they will give you more details on the schedule and the criteria for the design.

MR. MILLUNZI: The handouts that are coming around, we have three-ring binders that you will be able to put everyone's presentation in today so that when you leave it will be in one package.

I would like to say good morning and to thank you for this opportunity to meet with you today to brief you on the status of the DOE HTGR program.

It has been almost a year since we last met with you on this subject. I think the last time was February 5th, 1985.

My name is Andrew C. Millunzi. I am the manager of the safety and licensing for the advanced reactors at DOE, both the HTGRs and the LMRs.

Today, of course, we are going to be talking on

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the HTGR.

Before I get started, I would like to comment, we were very interested in hearing your comments that you were exchanging at the beginning of this meeting. I think you will find that we have addressed every one of the items that you mentioned, and we will continue to address them, starting with the number of interactions.

I think when you look at our detailed schedule, we fully agree that it is very appropriate and helpful for us to have good contact with the NRC so that people can understand the design that we are asking to get a license for.

What you will find as we approach that, we want to make sure that you understand the design so that you can pass judgment on the licensibility. However, we will talk later about how we would like to have that interaction proceed.

One of our concerns is the roles and responsibilities of an authority between both the applicant and the regulation, and we would like to talk about that a little bit later.

Relative to the need of a containment and separation of items such as Dr. Siess mentioned, we agreed with that. Our approach, as you will see when we come to it, we do not believe in attacking things by labels. We

are driven by requirements, and I think what we see is what is the function that containment would have to perform? How good does it have to get done, and how many ways could you achieve that?

I think you will see that we are attacking that problem because it is as important as Dr. Siess pointed out.

As far as interacting with the ACRS, we will be most anxious to have the interactions. Our TBD up on the schedule that Tom King put up, we were hoping we could put some dates on it as well as he did. So we would be anxious to enter into that discussion with you at whatever time is appropriate.

First of all, getting back to our presentation, I really appreciate your agreement on the agenda. We recognize that most people are very anxious to get at understanding the design.

In reviewing some of the past problems, we have come to the conclusion that people have jumped to that step too soon.

By that I mean we believe that it is vital to the review of a design that this review be completed with an understanding from the top down of what the approach to design and licensing was, what the requirements that the end product is expected to meet, and how these requirements are

to be met by the design, construction, and operation of the plant.

So we think it is absolutely vital that you understand these requirements, so that when we get the design you can understand how that design evolved.

A key point that we will be making over and over again, what we all have to do is assure that we meet requirements.

There are many ways to meet the requirements. We need to draw upon the creativity of the design organizations, the operating and the constructing operations to be able to do that in the most efficient manner.

Therefore, what has to happen is they must be given criteria. Then other people; namely, NRC, must be evaluating how well those criteria are met by the way the applicant is proposing.

The next job -- and it should not be in the mode of telling them how to do it better -- next of all, then, it is necessary to monitor them to make sure they are doing everything they said the way they said they were going to.

So with that, our approach to this first briefing with you and for subsequent ones will be from the top down.

(Slide.)

We will always be relating things back to the requirements and what it is one is supposed to accomplish. For this meeting, the objectives of this briefing with you are to brief you on our approach to the design. We would like to brief you on the licensing approach and methodology which has been proposed. We would then like to brief you on the ACRS and brief you on the design status, and we would like to receive your comments, of course.

We're most interested in your comments on our proposed licensing approach and methodology.

A fourth bullet that I've inadvertently left off there, but it's most important, and I would like to read it to you. The fourth bullet is, one of our objectives is to demonstrate that this concept is being developed in a disciplined, comprehensive, cohesive manner by a well-integrated capable team. We know that we can't expect you to give approval of the safety that you hear, but we fully expect that when we're through, that you will begin to appreciate the capabilities of this concept meet the safety and licensing requirements.

(Slide.)

I'd like to go back and review our program objectives at DOE, and there are these which we have presented to you in the paste.

(Slide.)

The highly disciplined, capable, cohesive team that I talked to you about is composed of these organizations. What we have here is a unique combination from the very outset, with a lot of interaction of the end user, the developers and the vendors, as you can see from this list

DR. SIESS: Where are the end users on there?

MR. MILLUNZI: The Gas-Cooled Reactor Associates.

DR. SIESS: Does that constitute your constituency of possible users?

MR. MILLUNZI: That is as of the present time, Dr. Siess. Yes. This is the most visible one who is providing support to the concepts. About 30 percent of the user utilities represent about 30 percent of the generating capacity of the country.

DR. SIESS: Am I correct that GCRA has not dropped its interest in the other forms of HTGRs and is concentrating on the modular?

MR. MILLUNZI: I can't speak for HTGRs, to the extent of having dropping. I think I can comment on the word "concentrate," and they certainly have concentrated. That is by far their main focal point. I think they would always continue to look, but as of right now, for our

program with the GCRA Management Committee agreement, this is the primary focus to make this reference concept applicable. This team then, as was discussed with you last year, completed a comprehensive, disciplined evaluation program that resulted in the selection of a reference concept on which we will focus all our efforts.

We started out with over 16 concepts and came down through this process to one, and that reference concept is a 4 x 350 megawatt thermal HTGR plant that has an annular core of prismatic fuel in a steel vessel.

(Slide.)

A picture of this reference concept is here.

I will not spend any more time on this, because we will be presenting this in overview fashion very shortly and in great detail during the closed session this afternoon.

(Slide.)

Next I would like to show you the overall schedule of our activities to augment the schedule that Tom King gave you. This schedule is the schedule that is contained in the approved licensing plan that we have submitted and received and approval received from NRC.

That licensing plan is broken into these four areas: a procedural approach, a technical approach, a design technology familiarization and design technology

review.

We have submitted the lciensing plan, and it has been approved, and we're working with the Commission Staff on this.

I'd like to comment, it has been a very, very fruitful interaction that's going on from our standpoint and myself personally, I really commend Mr. Dircks for establishing that Advanced Reactor Group. They have been most useful, and I think we're really looking forward to working with it. I think it's really going to help the process shorten up the time that we can get a license for this.

On the technical approach, we have submitted our proposed top level criteria, and we're anticipating getting NRC response next month. We would be reaching, we expect, towards the middle of the year, agreement on the licensing bases. We have sufficient resources right now in the budget, and we have taken into account Gramm-Rudman effects in March. We will submit the PSID in the content and quality that we had expected to at the end of September. We also anticipate carrying out activities to be able to complete these two milestones and get the licensability statement in the end of fiscal '87.

So even more so, I think, not only is it that the Advanced Reactors Group at NRC is doing a very commendable

job in being thorough, but I think the whole interaction would suffer if that didn't continue.

(Slide.)

Here I'm going to have a series now to show you the extent of the interactions with NRC. Here these are. I will not read them. They're all in the handouts. But you can see here on the procedural approach we're starting with getting agreement on what the licensing plan will be and taking into account the policy kinds of questions.

(Slide.)

On the technical approach interaction, we have submitted the top level criteria. We will be talking to you more about what we mean by this term "bridging." We have completed a method on selecting accidents, and I think this will go a long way today to beginning to answer some of the questions that Dr. Siess had brought up.

We agree that you need to do this, but we think before you get engaged in that, you have to have an agreed-to selection method, and that's why we want to talk to you about that, before we get in too far.

Also we think it's important to look at this question of safety-class selection. I think one of the problems that the industry probably has is the use of that term. We will be talking more about that as time goes on.

Also we are developing principal design

criteria. The principal design criteria are not the same, but they're akin to what people have done in the lightwater reactor area on the general design criteria, and we intend to get back to the original intent of the general design criteria, not to be confused with the general design criteria, because they contain such a high level of prescriptiveness and really aren't criteria. We have changed the name, not to get confused.

And of course, we're here today with the briefing to you.

(Slide.)

On the design technical issues, here is the item here. There is just a brief discrepancy between this schedule and the one that Tom put it that we have interchanged the decay heat removal and the fuel meetings. But here is a list. Our whole program. And the team that is back here. I have representatives from every team member here. All our plans, detailed down to the month and day are to provide these documents and meet these commitments. We've even identified the dates inside these months that they will be met.

The point I would like to make here is that the use of the word "issue" and how we will be addressing them. These items here all relate to functions that the plant has to perform in order to meet the requirements. We will not

be talking about these, as we go on, in what I might call the classical sense of talking about any one of these things in isolation, because it's improper, in our minds to treat them in isolation. They are an integral part of an integrated plant, but we will be going and addressing all these items.

(Slide.)

Next on the design and technology familiarization, we will be submitting a technology plan. I look forward to your being impressed with the manner in which that technology plan is being developed. We will be performing the PRA. I might add here that PRA will be used in this program extensively; however, we are going to employ the broad definition of PRA. It is unfortunate that PRA, in many people's minds, has been limited to just being a risk assessment item. Reliability engineering is what we will be utilizing in our activities to a great extent, and a PRA is a subset of using reliability engineering to assure yourself that you have a quality product to meet the requirements.

The PSID, we have submitted the outline and agreed to it with the Staff. The full submittal will be here. We will have a review of the preliminary safety evaluation report next June and the licensability statement, we're expecting next year.

The comments that were made about interacting

with the Staff of ACRS, we will be most anxious and very willing to work out an aggressive schedule of those interactions to assure that all of us can meet our respective responsibilities by this date.

That concludes the introduction on the part of my presentation.

DR. SIESS: Mr. Millunzi, you've emphasized the integrated approach, and you've emphasized a number of differences from current practice. You mentioned the principal design critcron versus the general design criteria, the different approach to the PRA. I suspect these are going to be areas where it's going to take some effort to get the ACRS full Committee familiar enough with the differences to be able to follow this thing. This thing will not be coming in in the conventional framework. We're going to have to educate people to the differences, and that's something that's just going to take a little more time. I just mention that.

MR. MILLUNZI: Dr. Siess, I think the English language, or my command of the English language is limited, and the choice of words, I couldn't find one which was less ambiguous. What I think you'll find is that we perform our work in a very disciplined fashion and try to make sure that people are articulating exactly what they're talking about. I do not believe that we will be doing things differently

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to the degree that it sounds. What we will be doing differently is that we will be clearly understanding between all of us what is the standard for acceptability and how did someone get there. That will be very clear, and it will be very easy for people to find. We will know what has to be done and how it has to be done.

DR. SIESS: But to get a 14- or 15-man ACRS to have an equally clear picture is not something you can do in an hour.

MR. MILLUNZI: Right.

DR. SIESS: The Subcommittee may be able to help you because of our familiarity with the other people on the committee and their interests and concerns, but eventually, it's going to be 14 people that have to be brought up to speed. I'm not pessimistic, just ralistic.

DR. CARBON: Another question. Sometime today, someone will be covering the difference in detail between a top-level criterion and a principal design one, and we'll be discussing what the technology plan is; is that correct?

MR. MILLUNZI: No. Later on today, we will not. We are scheduled in there, as you can see, to present the principal design criteria and submit them to the Commission. We're not prepared to talk about those today. We'll be more than happy to discuss them.

DR. CARBON: It wasn't so much to talk about them

in detail, but rather to explain a little more what the difference is.

MR. MILLUNZI: I think what the differences are --

DR. CARBON: Will that be covered today?

MR. MILLUNZI: I can cover that right now and tell you, in general, what they are.

The principal design criteria in the LWRs, in many aspects, are very prescriptive. They're not criteria. For some reason, the general design criteria have drifted away from their original intent, and what we have tried to do is go back to be consistent with the original intent of the general design criteria, which is, you provide general design criteria for a designer to meet, but you do not get to be specific in there as to actually defining that criteria.

DR. SIESS: At what point would you do that -- be specific?

MR. MILLUNZI: At what time will be?

DR. SIESS: How many principal design criteria can you have? Less than 58?

MR. MILLUNZI: The reason we're not prepared to do it, we're in the process of completing it.

DR. SIESS: And you're working with the Staff?

MR. MILLUNZI: Right.

DR. CARBON: But to a first approximation, the principal design criteria compare with the general design criteria.

MR. MILLUNZI: Right.

DR. CARBON: And the top-level criteria, what would the be, briefly?

MR. MILLUNZI: If you would permit me, I'll come to that in a little while. And we have a presentation on the top level criteria. Archie Kelly will handle that.

DR. CARBON: And the definition of the technology plan?

MR. MILLUNZI: I don't understand the question on the definition of the technology plan.

DR. CARBON: What is the technology plan meant to be?

MR. MILLUNZI: That is the plan of technology efforts that will have to be performed to supplement the technology that's not currently available to substantiate the conclusions that we reach in the license.

DR. SIESS: Mr. Millunzi, another question, I guess on semantics.

I have heard the word "discipline," although it didn't say a "disciplined engineering approach," which I have heard frequently if not recently.

What I didn't hear were the words "defense in depth."

Is that part of your philosophy?

MR. MILLUNZI: I will cover that in my next presentation. I am talking about discipline, and I seem to have misplaced a viewgraph here. Ah, I have it. I will talk to that.

We believe in defense in depth. I think the next part of the agenda was to get to our design and licensing approach.

(Slide.)

This viewgraph depicts very completely our design and licensing approach.

The blue part of this represents our approach to designing a safe, reliable, economical HTGR.

The yellow represents the methodology by which we will use to develop licensing bases unique for the HTGR in a format that the Commission is familiar with dealing with.

So again we will be approaching and defining our end product, which will safe, economic, and reliable, and

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develop a methodology which will develop the licensing basis unique for the HTGR in a form that the Commission is presently accustomed to working with.

This is in recognition to the question you were raising, Dr. Siess, about how much reeducating, et cetera, or changing.

Ideally, we would hope that eventually one of the benefits from an advanced reactor program is that this kind of an approach, which in reality is what all of us are doing, would be adopted, but we are dealing with the world as it is and not what we wish it to be.

Therefore, recognizing that problem that you are having, it is important for us, incumbent upon us to develop this methodology, to make this bridge. Our presentation today will follow this.

Archie Kelley will present from the top down -- we are working this for you. He will present the top level regulatory criteria proposed and the user requirements that we use with the integrated approach.

When we are through, you will find that the integrated approach is nothing magic. In fact, if we are successful in our interactions with you, when you are done you will say he didn't tell me anything new. It is not new.

All the integrated approach is, is a very

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disciplined way of describing how each of us does our job that we feel we are doing today. That is why the plants out there today are safe.

Our plant here is going to be equivalently safe. All of our comments and our need for developing this plant have nothing to do with the level of safety in the present operating plants. We believe they are safe for the American public to use.

Then Tony Neylan and Bill Sherden -- Neylan is from GA, and Bill Sherden is from Stone & Webster -- will describe how the NSSS vendors and the architect engineers take these top level requirements using this framework, developing the engineering product.

Fred Silady then will say -- with this product, he will describe the methodology that we intend to make this bridge to be able to develop these unique licensing bases for HTGRs in a format that NRC is familiar with.

I am a firm believer of defense in depth. Maybe it is because all these years I keep telling somebody I am 19 years old and all of a sudden one of these days I am going to realize I am really not.

That is why these plants are safe. They have been built in a disciplined way and with defense in depth.

(Slide.)

That is why nuclear power -- I am a firm

believer in it. So we believe in defense in depth.

We believe we are following it in a very, very good way, which is going to provide excellent defense in depth.

The way we do it is in pursuit of four goals:

First, our first goal is to maintain that plant in safe operation.

Next, because we are dealing with machinery and with people, we are providing defense, that in the low probability that you aren't able to maintain this safe operation, we are able to protect the plant.

We need to protect that plant because that is the investment part, and we have laid, as you will see, a very high requirement on how good we should do this job.

Then, consistent with nuclear energy, being the only one who moves out and looks at the consequences of extremely low probability events happening in their industry, we do maintain control of the release of radionuclides, and we are going to do that, as you will see, in a very unique way.

Next, another aspect of nuclear power which is unique to any industry in this country, we maintain emergency preparedness in the event that in the extremely low probability that we are unsuccessful at this level of defense.

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Now, goal one is to be achieved by the highly reliable operation of the plant with well-trained personnel. This will be made possible by providing a well-conceived design that is constructed to high quality control and assurance standards. That is number one. That is how we intend to get here.

We intend to use those high quality standards of quality control and assurance in goal two and in goal three to augment, that we will provide defense by utilizing the inherent characteristics of the items in that plant and by utilizing passive safety features.

We intend to do these jobs so well that we will place minimal reliance on four. We will have an emergency preparedness plan, but that plan will not call for sheltering or evacuating from the general public after 425 meters from the plant.

Therefore, we have imposed upon ourselves that we maintain control of the release of radionuclides to the extent that any neighbor living more than 425 meters from that plant his daily activities would not be affected and he would not have to be evacuated or sheltered.

We believe all the things that are necessary to achieve these are going to give us another type of safe nuclear power plant. In my mind, it is another type of nuclear power plant.

The safety of nuclear power plants isn't limited by the coolant. It is how well you do these jobs.

DR. SIESS: Am I correct that in that last objective you intend not only that the person near the plant be protected from harm but that also he be protected from fright?

MR. MILLUNZI: Right. And also, going along with that, I almost fell into the trap that I accuse other people of, of being too complacent about another important item.

We from the very beginning are stressing operations and maintenance, which result in putting in safety in the beginning, and we are also making sure that we do a superb job on protecting the personnel who are working in the plant.

DR. MARK: I really like the kind of things you are saying. I am a little uneasy perhaps of the enormous amount of having things come out the way you describe them. It depends upon the management of the senior personnel of the organization that is involved.

In what way and to what degree and by what means do you think you can ensure that the people handling things are handling them correctly?

MR. MILLUNZI: I think, as you will see, if I can put this defense in depth back up here, we intend -- and as I said earlier it is incumbent that when you go through

this that you have a full team from the very beginning and that they be capable and that you have the end user, the guy who is going to build and construct it and who is going to operate and maintain and be involved in the program from the very beginning.

We have gone to -- we think we have on this team very, very capable people and capable organizations. We will, as you will see throughout the development of this program -- as you recall, when we submitted the response to Public Law 96-567, which was weighed to recommendations on how to improve the safety of LWRs, which I was responsible for, our number one requirement there was related to personnel, and we will be paying extreme attention to the training of everybody involved from the management down and their qualifications.

The plant is not going to have a safety problem until it goes into operation and you have machines and people, and it is necessary, for example, that the personnel, the maintenance and the operating people, all part of Big O as we call it, Big Operations, are well-trained and qualified.

So we will be paying close attention to that. We will be trying to select, recruit, and keep people of high quality.

That is why I used those words in my introduction

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so carefully. We really are dedicated to that. People are going to make this thing work.

DR. SIESS: Will this design have the same slow response characteristics that they have, for example, at Fort St. Vrain?

MR. MILLUNZI: Yes, and that, Dr. Siess, is right here. We are going to use the inherent characteristics of graphite and of fuel along with passive features. We are looking for that all the time.

But that is right, we want to build on those kind of characteristics. What we are trying to do is provide time -- you know, one of the strategies here is to provide time for the operations, the Big O, to make a decision and take action.

Now, you can't stop with that statement right there. You have to define what actions and what decisions they have to make. And as you will see, as we develop that we are concentrating on that.

DR. SIESS: Stepping ahead about three steps, when the license application comes in for one of these, can we assume that a training simulator will be an essential part of it?

MR. MILLUNZI: Most of the guys on the team will die laughing right now when you ask me that question. I have a number of red flag words, and one of them is

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"simulator." I don't know what the hell anybody means when they say "training simulator."

I will tell you what we will do. We will utilize the latest technology in human engineering -- which is another term I don't like but it is one that people are familiar with -- we will use the latest techniques in assuring ourselves that we get training. If it means using some computer -- whatever it takes -- we will.

The word "simulator" bothers me because it means different things to different people. But I think we do need to have a device which is good for training people to identify and see how they would respond to things, help them learn how they should respond to events, et cetera.

DR. SIESS: I don't understand why it means different things to different people. I thought NRC had a reg guide out that pretty well defined what a simulator was and what it had to be able to do.

MR. MILLUNZI: I know, and I hope that as we go on we can improve on that definition.

DR. CARBON: Andy, you are talking about high quality people, highly trained, and so on, running the plants, but you aren't going to have any say-so over that. That will depend on the company out there that owns and operates it.

It seems to me you almost ought to be aiming

toward designing a plant that you can have any dumb old guy out there operating.

MR. MILLUNZI: Right. Well, one of the features that you will see, an objective of the program, is to have a plant which is fully automated. You will see that our staffing goals are to reduce the staff.

We would like to get to the point -- I am in agreement with you -- and our intent as a start to that is to have a fully automated plant and to use, even with those people, though -- I think it is incumbent for us to use things -- like I am in complete agreement with Chet on using things like the simulator, I think, as he has it in his mind that it should be, to train them. You still have to train the people.

So an important part of our program is that we have the utilities involved from the beginning, and one of the tenets that we are trying to develop in this whole program and the attitude that we are trying to develop in this whole program is remember you are going to have to train those people.

I am probably -- besides Dennis Wilkinson -- probably the biggest booster for INPO. We stress the importance of INPO, the kinds of things that INPO is trying to achieve, and the DOE program will try to lay down the framework and the emphasis, and hopefully through that, in

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cooperation with the NRC, we can develop that attitude.

That is why it is important, when we get into this, that you will hear us talking about we are beseeching you because we will be highly resistant to prescriptiveness.

We need to get the people to understand that it is their responsibility to design that plant right. It is their responsibility to operate and to maintain it, and as part of that fulfillment they must understand that plant. They must be responsible for choosing the options and getting them approved. They cannot rely on anyone else, the NRC or DOE, to tell them.

In the beginning, we should guide, but there is a difference between guiding and telling. A problem that has crept in, too, is that the guidance has become prescription. We want to get back to the guidance that relates to the criteria, for example.

So in answer to your question, we are striving mightily. The hidden agenda in this is to train the whole industry that they need that, and I would like to add that in my personal opinion it is not limited to the owner-operators. The same thing applies to all of us.

So that is how we are attacking it, and I don't know of any other way to do it. In the end they are responsible. Therefore, we have to work together to make

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sure they are capable of fulfilling that responsibility.

I think they are responsible now, looking at the great safety record for nuclear power plants. However, when one looks at improving -- my definition and the reason I use that word is not that improvement means what is there isn't good enough. My thing is all of us can be improved, and that is what we are looking for -- is for improvement.

And I am personally -- the number one item is personnel.

DR. SIESS: Max, I'm not sure I agree with you that I want a plant that any old guy can operate. I have a feeling I'd be more worried about complacency than I would about incompetence.

DR. CARBON: My point was, DOE doesn't have any control over the people.

DR. MARK: I'm still left a little bit, not really uneasy, but if there was such a thing as a sleazy management, then all the things you're saying, don't necessary apply, and I don't see any way of assuring that we don't run into a sleazy management.

MR. MILLUNZI: Along that line, Carson, we are trying to build a just reward system, okayt? And it goes the following way:

This plant, by what we're going to do in 3 and 4, will be such that anybody 425 meters from that plant, will not be injured. The guy is sleazy is going to lose his shirt, because his plant is down and out.

So that's our ace in the hole against that kind of thing.

DR. CARBON: I don't want to prolong this, but I thought, that's just contrary to what you said in Item No. 2.

MR. MILLUNZI: No, Item No 2 here is here, if I could, I'd like to defer that to we're all done.

DR. CARBON: Maybe we'd better, because I think we're falling pretty far behind schedule.

(Slide.)

MR. MILLUNZI: Lastly, in the summary, then, what we're doing is, we are identifying the criteria, and that is done. We're awaiting the final approval. We are developing the process to derive the licensing bases unique for this, which NRC is familiar with. We have briefed and met with the Staff, and we will be submitting our documentation in the next several weeks. And then we take this an apply the process to identify these licensing bases, and these are in process.

Archie Kelley will follow me. He will be talking about these criteria. Then Tony Neylan and Bill Sheridan and I will talk to you about this approach. Neylan and Sheridan will talk about the product that results from that, and then Fred Silady, with his presentation, will cover points 2 and 3.

So with that, I'd like to turn it over to Archie Kelley from the Gas-Cooled Reactor Associates.

MR. KELLEY: Good morning. My name is Archie Kelley with Gas-Cooled Reactor Associates.

As Andy told you, Gas-Cooled Reactor Associates represents roughly one-third of the nation's generating capacity. Our members are vitally interested in the

development and support the development of the gas-cooled technology in this country. Also to confirm what Andy told you, our surveys of our member utilities, as well as potential process heat users, indicates that this concept you're going to hear about today, indeed, is an excellent match to our utilities projections of their future energy needs in the next several decades.

In addition, we think it's an even better match than the former large plants we worked on for the potential process heat and higher temperature markets which may be out there in the future for nuclear power applications.

To move into my topic today, as Andy told you, we're going to use this as our road map for the presentation today.

My subject matter is to brief yo on the top level user and top level regulatory criteria that we feel appropriately fill these top two boxes on the left-hand side of this diagram.

(Slide.)

I'm going to do this in three parts. First, very briefly, the purpose of these requirements, some of the ground rules by which we develop them.

Secondly, I'll tell you what the utility user requirements are that are driving this design you're going to see today, and then finally, most importantly, describe

to you the regulatory criteria that we have proposed to the NRC Staff to use to evaluate our conceptual design. And in doing that, we came up with some bases for selection which I'll identify and then the proposed criteria themselves.

(Slide.)

First of all, with regard to purpose. The purpose of these top-level criteria and requirements, quite simply, is to help us define how safe and how economical we want the power from this plant to be.

As Andy has told you, working from the top down, we have defined four goals that we will allocate these requirements and criteria to. Goal 1 being maintain safe plant operation. Maintain plant protection is Goal 2. Maintain control of radionuclide release, Goal 3. And finally, maintaining emergency preparedness.

So our goals, you will see, are allocated against -- excuse me. Our criteria are allocated against the structure of four goals, which we believe are those goals which are necessary to maintain safe economical power.

(Slide.)

Moving then into the second topic, the utility user criteria. Here, of course, I will be emphasizing the aspects of economics, because that's primarily what the user is focused on. You will see, of course, that there was a crossover by the user. He is also interested in the safety

aspects of this plant, but my emphasis from the user viewpoint will primarily focus on the question of how economic, beginning with some definitions from an overall viewpoint of what economic means to our utilities. We have some numerical goals that have been defined, first of all. We have a target that we show a 10 percent economic advantage over state-of-the-art coal plants. We consider that the appropriate competition for this plant at this time.

We defined a siting envelope we want this economic criteria to be met, considering a siting envelope which cover roughly 85 percent of U.S. sites and other criteria, such as the service life of 40 years, and such things as would cover the overall plant.

Moving then into the criteria for maintaining safe plant operation, the overall criteria from the user viewpoint is that we show an equivalent unavailability owing to planned outages of less than 10 percent.

You will recognize that to meet this, there are a number of suballocations that relate to such things as how we maintain and operate this plant, and there are safety considerations with regard to the worker exposure, and so forth, in order that we might meet that target. And those sublevel criteria are, in fact, identified by the user.

DR. SIESS: Excuse me. In that economic

advantage over coal, what does that assume about the back end of the fuel cycle?

MR. KELLEY: This does not assume, obviously, the recycle fuel cycle.

DR. SIESS: No recycle?

MR. KELLEY: No recycle.

DR. SIESS: And permanent disposal of the 1 mil per kilowatt hour?

MR. KELLEY: That's correct. We have a set of user defined economic ground rules, and we're evaluating that. And that is an example of one of the elements in the ground rules.

DR. SIESS: I just wanted to make sure. It wasn't really pertinent to this.

(Slide.)

MR. KELLEY: In the area of the second goal, which is maintain plant protection, there's a lot of user interest, and therefore, we have a number of requirements in this area that are imposed on the design. Some of the key ones I've shown up here.

First of all, in terms of maintaining plant protection, the user is concerned about those events that, in terms of equipment damage, may be relatively trivial, but may result in cumulative plant outages that impact to the overall availability. We therefore have a goal that the

equivalent unavailability, owing to such unplanned equipment outages be less than 10 percent.

You can take this 10 percent and add it to the 10 percent planned outage criteria, which I showed you previously, and we have an overall goal that the availability be greater than 80 percent, the equivalent availability.

Moving down, in terms of level of concern, for more serious events, you are obviously concerned about equipment damage. We therefore had a criterion basically that the annual expected value of damage to equipment be less than the insurance premium which we use in our cost estimate of $4.5 million a year. Obviously, we're not talking about the expectation of an annual -- in fact, an annual loss of $4.5 million, but overall events that might impact equipment in the plant, we're asking that on the average, they be shown to be less than $4.5 million.

Finally, moving down to even more serious potential events in the plant, those, in particular, might be of nuclear significance and might have public repercussions. The user, of course, is concerned, additionally, not just because it might impact his plant, but because, in fact, it might impact the entire base of installed HTGRs in a population of HTGRs in the future. Therefore, there's obviously quite an aversion to such

events. That's shown by our criteria that we require, that the designers show that there be a mean likelihood of a loss of a single reactor, be shown to be less than 10 to the minus 5th per year.

DR. CARBON: This is total loss? Wipeout? I mean financially and operability?

MR. KELLEY: This effectively would be a loss of reactor so serious that it would be a write-off, as far as the utility was concerned.

DR. SIESS: A write-off of one reactor out of four?

MR. KELLEY: That's right.

DR. SIESS: But not such that it would disable the whole plant?

MR. KELLEY: The concern is obviously, if this were a nuclear-related event, again, that it could have public repercussions, that even though the other plants are not damaged, they could be commonly affected.

(Slide.)

Finally, as Andy has already told you, we do have a key user requirement in Goal 3, relating to maintaining control of radionuclide release. The user is requiring that we do this so well with this plant, that we would, indeed, meet all the top level regulatory criteria without taking credit for sheltering or evacuation of the public, such that

the utility user need not plan for the offsite evacuation and sheltering of the public. This currently has been interpreted, and we believe quite conservatively in the design process as meaning that we show that we meet the protective action guide doses for sheltering. Mainly, 1 rem whole body and 5 rem thyroid for events that have assessed mean frequences of greater than 5 x 10 to the minus 7 per year.

DR. CARBON: What was that again? 1 rem whole body and 5 rem --

MR. KELLEY: 1 rem whole body, 5 rem thyroid.

DR. SIESS: That last statement, is that equivalent to saying that the probability of exceeding the PAG is 5 x 10 to the minus 7 per year for all events?

MR. KELLEY: It would be less than 5 x 10 to the minus 7 per year for all events.

DR. SIESS: Okay, for all events. Okay.

How did you arrive at the 5 x 10 to the minus 7? Working backwards from the safety goal?

MR. KELLEY: In effect, you'll see further discussion of this in Fred Silady's paper, but in effect, we did work back from the safety goals.

DR. SIESS: Is the 10 to the minus 5 then equivalent to a core melt criterion?

MR. KELLEY: Yes, but again, it's from the

viewpoint of equipment damage that would have a nuclear consequence. Core melt doesn't have much of a meaning for a graphite core.

DR. SIESS: I said equivalent to that.

MR. KELLEY: In loose terms; yes.

DR. SIESS: So it would damage the plant severely.

MR. KELLEY: And have offsite public repercussions; yes.

DR. SIESS: But not offsite doses?

MR. KELLEY: But not significant offsite doses. Again, what we would be concerned with is events that would cause the public concern, even though they would not actually necessarily evolve health effects.

MR. MILLUNZI: I might add, Archie, in order to help the situation, the design also has to be adverse to long outages, six months.

MR. KELLEY: That's a point. You're not seeing all the tiers of requirements that we have, but in terms of the 10 percent unplanned outage, the user has an aversion to very long outages, namely, those greater than six months. They can only contribute 10 percent to that 10 percent.

DR. SIESS: That would be something like Davis-Besse?

MR. KELLEY: Yes; that type of thing.

DR. SIESS: You would expect that something like that would affect all four units; right?

It would be a plant shutdown, not a reactor shutdown?

MR. KELLEY: That's the concern for the 10 to the minus 5th criterion.

MR. MILLUNZI: That's a long outage. It's per reactor.

DR. SIESS: Well, I would be willing to bet that if you had something that the NRC considered like Davis-Besse, it would shut down the plant.

MR. MILLUNZI: That's right; yes.

DR. SIESS: We're talking about a regular shutdown now.

(Slide.)

MR. KELLEY: Let me move into my last topic, the most important one here, and discuss with you the approach that we proposed the NRC Staff with regard to top level regulatory criteria for this concept. Obviously, it's not our role nor were we really interested in proposing unique or new top level regulatory criteria for this concept. But we were concerned that in the current body of regulatory criteria, Code of Federal Regulations, et cetera, that were a number of criteria that were rather unique to lightwater reactors, somewhat restrictive in terms of lightwater

reactor design features, a we felt it important to identify those criteria that were clearly generic versus those that appeared to have a lightwater reactor history to them or lightwater design background to them.

To do this, we established some selection bases, which I've shown on this Vugraph, and we hope that these selection bases would help us in this weeding process to identify criteria that, indeed, were truly generic.

The first basis for selection that we used were that the criteria must be direct statements of acceptable consequences or risks to the public or environment.

We believe clearly that requirement was independent of the design.

Secondly, quite clearly, that the criteria should be independent of the plant design. It was mentioning specific plant design features, and so forth. Obviously, that would not be the case.

Thirdly, in order to be useful to us in our design process, we believe the criteria must be quantifiable. Just as I've shown you, the user criteria were quantified. We believe the regulatory criteria used in this conceptual design process need to be quantifiable, in order to provide appropriate guidance to the designer.

So with these bases, what we did, we scre ned the requirements and criteria and guidelines that we find in the

regulatory documents and then identified those that we thought, in fact, were generic.

(Slide.)

Starting then again with the statement of overall safe economic power, we felt, obviously, the best candidate for this was the guideline criteria found in NUREG 0880, namely, the interim safety goals. Namely, there, the criteria that met our selection bases were the individual and societal mortality risks that are identified therein and also the cost-benefit criteria, which are identified therein, that would be invoked only if the mortality risk criteria were not met.

Obviously, we intend to meet these criteria, and therefore, based on the interim safety goals as currently stated, we would not need to invoke cost-benefit criteria.

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DR. SIESS: Are you familiar with some of the proposals for final safety goals?

MR. KELLEY: We have been following that, obviously, with great interest. Once those are promulgated, we will obviously have to consider those.

I might note, one thing you will see conspicuously absent here in NUREG-0880 is -- of course, there is an identification for a design performance goal for large scale core melt. However, we felt that that did not in fact meet any of the three selection bases which I have identified to you previously. It is not a direct statement related to public health and safety.

We don't believe it really is independent of plant design. We believe it is light water reactor only, and it is in terms of understanding the consequences of core melts and light water reactors. And finally, because of the terminology, it is not really something we can quantify on a graphite reactor.

DR. SIESS: I agree with you 100 percent, but I think you have got some people to convince.

MR. KELLEY: Yes.

DR. SIESS: Not only have core melt criteria or core damage criteria begun to assume a primary role, but there is a move toward including descriptive containment performance criteria, which is a particular problem.

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MR. KELLEY: Yes.

Recognize, however, as I showed you previously, from a user perspective in any case we do in effect have a performance criteria related to damage to one of the reactors.

So obviously, that is not neglected in our design process. We just feel it is coming appropriately from the user.

DR. SIESS: But when you say a user requirement, it means you hope that it wouldn't be an NRC requirement?

MR. KELLEY: No. We would hope it would be a user requirement.

(Slide.)

Going through each of the sublevel goals in terms of maintaining safe plant operation, we believe obviously the candidates there are the permissible dose levels and activity concentrations for unrestricted areas, which are contained in 10 CFR 20.

Analogously, the numerical dose guidelines contained in 10 CFR 50, Appendix I. In goal two, obviously that would apply as well. But there are obviously concerns related to occupational exposure criteria. So those would be invoked under goal two.

(Slide.)

Under goal three, related to maintaining control

of radionuclide release in the event of off-normal or accident conditions in the plant, we are invoking 10 CFR, Appendix I doses applied on an expected value basis to events which might occur in the plant lifetime.

In other words, by taking the sum of the product of the frequencies and the consequences, we show that we meet 10 CFR 50, Appendix I, offsite dose values.

For accident events, obviously we have invoked the numerical dose guidelines contained in 10 CFR 100.

Finally, with regard to the goal of maintaining emergency preparedness, we believe that the criteria should be the criteria contained in EPA-520, the PAG doses contained therein, which are guides in terms of triggering offsite actions such as sheltering and evacuation.

This completes my summary of the top level regulatory criteria.

I recognize that we have submitted a document to Tom King's group which identifies our proposed criteria in some detail. It also identifies the selection bases which I described to you as we understand it. This is currently being reviewed by the NRC, and we expect to hear from them shortly on that.

DR. SIESS: I have a question I forgot to ask. It may have been answered.

On one of the slides you said the siting

envelope covering 85 percent of U.S. sites.

Can I assume that that is primarily seismic?

MR. KELLEY: It covers a number of things -- seismic, wind loading, temperature. For example, Andy mentioned to you we have a criterion of 425 meter exclusion area boundary. That number actually came by enveloping 85 percent of the sites.

DR. SIESS: Existing sites?

MR. KELLEY: Existing sites in that case, that is correct, existing or sites under construction. 425 meters our assessment tells us bounds 85 percent of current sites.

DR. SIESS: But in terms of potential sites, the only thing to exclude would be certain seismic regions, right?

MR. KELLEY: Yes.

DR. SIESS: If you chose a state, you couldn't choose California?

MR. KELLEY: Certainly, there are parts of California which would be difficult.

DR. SIESS: And a few in Wyoming.

MR. KELLEY: Okay.

(Slide.)

Putting our road map back up on, then, the next topic.

Now that you know what the top level criteria and

user requirements are that have been defined, Andy Millunzi will describe to you the integrated approach technique we use in a system engineering sense to meld these requirements into a design.

MR. MILLUNZI: I am going to cover, as Archie pointed out, a description and a discussion on the integrated approach.

First of all, when one has that, if you are going to set out to do a job like this you have to know what are the purposes and benefits of doing this.

(Slide.)

The purpose of the integrated approach is used to develop requirements. It is used to evaluate the options that one can develop to meet the options, and it is to communicate with reviewers.

By this, I mean that the first point for a nuclear power plant, requirements have to be developed which will respond to the top level user and regulatory requirements.

So taking those requirements with the integrated approach, the purpose of it is to allocate, to develop and allocate those requirements. Then it is to evaluate the options that the design, construction, and operation institutions propose to meet those requirements.

And then a purpose for it is to communicate with

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all kinds of reviewers. It does not mean the reviewers at NRC. It is not restricted to anybody. It is to be able to communicate what this plant is doing and why it has its capabilities.

(Slide.)

The benefits from the integrated approach are many.

In thinking about our presentation last night, I have added a few viewgraphs which aren't in your package, but I am sure we can get copies to you.

The benefits from it are that the development of this integrated plant model provides insight into the design structure of the plant in terms of definition of functions, definition of a minimal set of components which affect each function.

This will lead to a better understanding of the plant in that the exact reasons for having each piece of equipment are identified. That is a very important item, to understand why every piece of equipment is in there.

Every piece of equipment is in there to perform a function. This may assist in defining what is or is not functionally required to be safety related. It will give a definition of which components are most critical to plant operations in terms of economics and safety.

(Slide.)

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Continuing on, it provides insights into the importance of human interactions, which are knowledge-based behavior, rule of skill-based behavior, or institutional or administrative.

Three, it provides a framework for information processing techniques which allow the human to perform a top-down analysis of any plant condition, determine the plant status as expeditiously as possible, and thus reduce the time required to arrive at a decision and reducing the likelihood of making a wrong decision.

Back to our emphasis on how important the people are and their training.

(Slide.)

Continuing on, it provides an analysis which will indicate the importance of institutional contributors which can then be compared to their role in the existing plant function administration. This may lead to areas which may require emphasis or improvement.

Continually, you need to be operating these plant, to continually review yourself, learn from yourself, and how do you identify where these improvements can come from.

And five, it provides a functional model for an optimal offsite emergency preparedness which our emergency plan could be evaluated against.

(Slide.)

The benefits, continuing, the integrated approach is expected to minimize the total cost of power by having all of these attributes and having the interactions. We would have a shorter licensing time, a shorter construction time, be able to minimize retrofits and minimize operating and maintenance costs through the better understanding of things before and after the plant is put online.

(Slide.)

In the design areas, the savings are due to a clear understanding by the designers, contractors, and operators of what their roles and responsibilities are.

The bringing online and the operating of a plant requires many organizations and many subunits inside these organizations, and you need to pay attention to interfaces so that you don't have somebody building an engine for your airplane that is too light for the payload that it is going to take off. That is what happened to Westinghouse, for example, in that area.

And early identification of interfaces which reduces the risk of later more costly revisions. That is part of the point I think, Dr. Carbon, that you were talking to.

It gives a visibility of the basis for the design requirements. This is very, very important. It would help

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eliminate unjustifiable retrofits.

Lastly, we expect to get savings due to how it will affect our development programs.

Now, I will get into -- we needed this kind of a framework, we found. We needed something which is the integrator. So I am going to give you a very brief description of the integrated approach.

As Dr. Siess says, it will take some more interactions for people to become more familiar with it. However, you will find, as we talk to you today, and hopefully and assuredly as we get to talk in more detail, that we are not talking about anything very sophisticated. We are not talking about something that people don't do already.

The difference is we do it in a very disciplined, organized fashion. We articulate very clearly what it is we are doing, and we provide the traceability and the visibility.

So we have put in systems engineering way how everybody believes they are doing their job. Now to do this and to get all these benefits, we discovered you have to go back to square one. All of us have to go back and say from the top down, why are we here, what are we after?

(Slide.)

Our objective, all of us, is to use the heat from

a nuclear reaction to produce electricity safely, economically, and reliably. Therefore, the owner of the plant, if it wouldn't be for the radionuclides that his product possesses, he would only have to be concerned with meeting the economic criteria.

Because his product contains a toxic or radioactive material, society has imposed a regulatory body on him whose responsibility is to protect the health and safety of the public.

So the owner-operator of this plant has to now use a plant which he could sell at a profit electricity, but he also has to meet regulatory requirements.

Now, with that dual set of requirements he has to try to figure out how he is going to approach that. There are many ways that those requirements can be met.

As I said earlier, he has to be afforded the ability to develop the option as to how he wants to do it. The people who are lending him the money have to evaluate if it looks like it is economically viable, and the regulatory bodies have to see if it is adequately protecting the health and public safety.

But both of those reviewing organizations, then, have the responsibility to evaluate if his proposed way is good enough. They must resist telling him how to do it.

The way in the nuclear power business we have

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elected to do it is --

(Slide.)

-- we first of all develop a design and we construct and operate it so that it will stay in the normal operating envelope.

(Slide.)

Recognizing that this plant has people and equipment involved, we then provide features, both in operation and training, et cetera, so that we will maintain the plant protection.

(Slide.)

Then, knowing that we have this radionuclide content in the plant and we have to meet regulatory requirements, we in our obligation to society have to be able to -- in the event that the damage gets to be so severe that we are releasing radionuclides, we have to maintain control of those radionuclides so that they do not injure the health of the public.

(Slide.)

Then, last of all, the nuclear industry again, unlike any others, has to provide for emergency preparedness. It turns out that it is all of these activities that the utility, the designers, and everyone else has to perform adequately enough to meet these requirements.

There are many ways that this can be done, and so it is the responsibility of the applicant and his suppliers to develop a way to do that.

How we do that is nothing new. This is how all engineering jobs do it.

(Slide.)

What we have here is a goal. In order to achieve that goal, there are functions that have to be performed. In this viewgraph I only have one, but there are a number of functions which go together that have to be completed in order to achieve the goal.

As engineers, what you do in a systematic way is we take the function, and a function is modularized. You have subfunctions. For each of these subfunctions you have systems and components which, together with human actions, enable you during operation and maintenance to achieve the functional requirements of the subfunction which, taken together, enable you to meet the functional requirements of the function, and the functional requirements of all the functions accumulated enable you to meet the requirements of the goal.

So as you can see when you develop this, you have what we call a tree. So what we then perform is a functional analysis of this tree to try to determine, one, what are the functions, what are the subfunctions that have

to be performed, then define and allocate from the goals how good that has to be done. That is what functional requirements are.

The functional requirements at each level are then further allocated down to the subfunctions. It is against these requirements, then, using design criteria, that the systems and components get designed along with the humans to meet all of these items.

It is important that in many aspects right now the visibility and the traceability -- there is improvement that can be done in that. So if from the very beginning we are worrying about traceability, we have developed a documentation logic which will preserve for us at all times for everyone to see why did we do what we did.

(Slide.)

This is accomplished by having these sets of documents.

Starting from the top, there is an overall plant specification. In that document are user requirements, NRC requirements.

Taking all of these, we perform plant level analyses, trade studies, design studies, and out of this we define the system requirements.

These system requirements then are passed down to the system design description, the SDD. These become requirements.

The system designer then has the opportunity and the responsibility and the option to develop options as to how those requirements will be met. They do the same thing as what was done up at the top. They perform system or subsystem level analysis, trade studies, and design studies.

Out of that will fall out design selections. These design selections in turn are then passed down as requirements to the lower document.

For example, from the system, the system design selections are then, with their requirements, passed down to the subsystem as requirements. Those designers are charged with the responsibility of defining the optimum way to meet

those.

Finally, we get component design specifications which can go out.

Now, throughout all of this, because of the number of organizations that are necessary -- and even within the organizations, all the suborganizations are involved, and the interactions between all of these functions -- it is absolutely necessary on a systemwide basis to be able to have the multi-system interface analysis performed and identified.

And so, we go through and we have interface requirements documents which identify how and in which areas the interfaces are to take place, so that all of the organizations involved know what is required for them, and other people's requirements are taken into account.

For example, a heat transport design engineer cannot establish any requirements for himself. His job is to understand what the cooling requirements are for everybody else in the plant. Then he takes all of those requirements, develops a design envelope to meet all of those, and he makes sure that each of their requirements is met. He is an interface. The heat transport man is an interface. He has the interface with everybody in the plant.

The fuel designer must interface with everybody

in the plant.

The purpose for the scram rods in a plant is to be able to turn off the heat generation; that is, turn off the reactivity fast enough so the temperatures don't get up high enough to cause damage to the fuel.

Well, the performance requirements of the fuel must be translated to the control rod design engineer. It must be translated to the plant protection system designer.

So you need to worry about those interfaces, and that is what this will provide.

(Slide.)

To give an example, most people don't pay attention -- I shouldn't say that, but it is not explicitly recognized by many people. For example, in safe, economical power if you want to maintain operations there are four stages for the owner-operator which represent plant operation, and they are the ones which are listed here.

For illustrative purposes, we have taken to maintain energy production. This is the one that most people concentrate on. This is when the plant is producing power.

The functions at this stage, the functions that have to be performed, are described right here. All of these functions must be performed during energy production if we are to meet the functional requirements to meet that

80 percent availability number.

Therefore, we allocate among each of these what is their requirements. So, for example, everybody knows about having to produce reactor energy to maintain energy transfer and to convert that energy. But along with that there are many other considerations that have to go in, and they are here.

We perform a functional description for all of those and define what the functional requirements are. Now, none of this is any different than people do it right now.

Now, I would like to give you an example of how the design evolves.

(Slide.)

In this viewgraph, I am talking about how to get a goal one design, as we call it; that is, maintain normal operations.

The first thing we do is we develop the functions and requirements and make design selections to meet the goal one requirements to maintain plant normal operations during each one of those stages.

(Slide.)

Pictorially it looks like this. Here is goal one: maintain normal operations. We have requirements for that.

Those requirements then are allocated down to the

functions. Taking these requirements, then, we perform these analyses and trade studies. Based on this, we provide a design selection, and from that you get the design.

DR. CARBON: What is a trade study?

MR. MILLUNZI: For example, you can perform what kind of a vessel should we use in the circulator, for example. Should it be a water-cooled circulator, a magnetic circulator in the cooling?

All kinds of different studies.

DR. CARBON: You mean tradeoffs?

MR. MILLUNZI: Tradeoffs.

DR. CARBON: All right.

(Slide.)

MR. MILLUNZI: Then how do we proceed on to providing the design that not only can meet the goal one requirements but also now can meet the goal two requirements?

Okay, the way we do that, then, is you take that goal one design that you developed, which was focused on developing to meet the goal one requirements. Then we develop the goal two functions and requirements to maintain plant protection.

This is just independent. We just said, okay, our goal is to maintain plant protection. What are the requirements? What are the functions that we need to

perform to do that.

Then we evaluate the goal one design, the existing design, to see if they meet the goal two functions and requirements. If not, we develop the modifications or design changes.

(Slide.)

This is illustrated in this one here.

We start off. We have developed this plant design. Then we take a look at the goal two requirements and the functions and the requirements independently. Then we evaluate to see if these selections meet the goal two requirements, if that plant as it presently stands meets the goal two requirements.

If the answer is "yes," we then proceed to develop a design which will meet goal one, goal two, and goal three requirements.

If the answer is "no," then we have to determine what the additional functions and/or additional requirements are.

You go through the same analytical loop, but now you have to keep in mind that if I make a change to meet the goal two requirements, am I affecting its ability to meet the goal one requirements? You have got to have this integrated. You have to keep looking, balancing and trading off and keeping all requirements in mind.

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So do we go through this analysis with these design selections? That means that the plant will now meet both the goal one requirements and the goal two requirements, and now we take that design, repeat that process to see if it meets the goal three requirements.

I now come to what I always describe as my favorite viewgraph.

(Slide.)

It is this one here. On one viewgraph I think we have described the nuclear industry.

What we have up here is what has to be done.

What we have here is how we do it. When we put the functional requirements on here, we describe how well it has to be done.

Having identified the product and how we want to get there, we then turn to the four institutions to provide that product.

We have a design organization, a construction organization, an operations organization -- this is "o" -- and a maintenance. Big "O" is operations and maintenance.

It is important -- okay -- therefore that from the beginning all institutions must understand what is required from them from the outset.

Up here, the end-user has to understand and determine what it is he wants.

These people have to understand that is what he wants.

Now, the design organization is in here. He must understand when he is going to design a product which is goi to meet this requirement -- he has to have an understanding of how the constructor intends to build his design, and he must understand what the operations and how the operations intend to use his product.

Also, the constructor must know what the designer had in mind so that he can maintain that intent in his construction practices, and he also must know that that device that he built -- how will it affect the operations and maintenance.

You do not want to have a Chevette situation, where I build an engine and if I want to change the spark plug I have to pull out the whole engine.

So from the very beginning he has to know.

Also, the people who are going to operate and maintain it must understand how this guy's design, or the design he intended to meet these, and how he expected them to operate and maintain it. He also has to provide feedback so that the construction features and practices meet his.

Therefore, you have to bring all of these together in the beginning, and you have to integrate all of them, and this demands an integrated approach, and that is

exactly what we have.

We have the end-users, we have the NSSS, we have the constructors, and we are interacting with the operations. From the very start we are using this.

We provide a format which clearly defines what everybody is supposed to do. We have a format which defines what all the interfaces are, and we have that documentation logic which will maintain the visibility as people come and leave the program.

And that is a brief discussion of how we do it and what we do, and I will emphasize over and over again there is really nothing very sophisticated.

DR. CARBON: Fine.

(Slide.)

MR. MILLUNZI: I think now you have sat through all this approach and what it is, and I think it is appropriate at this time, I guess, for you to see what all of this has wrought.

And so, Bill Sheridan, from Stone & Webster, will start off talking about the overall plant. Tony Neylan will then follow with the NSSS.

DR. CARBON: Mr. Sheridan, we will need a break here pretty soon. Is it appropriate to interrupt yours?

MR. SHERIDAN: I think so. You mean to break?

DR. CARBON: If it is all right to interrupt it,

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go ahead for a while. If you had rather, we will take a break now and not interrupt you.

MR. SHERIDAN: I will take approximately 15 to 20 minutes, and Tony Neylan, who should follow in sequence with me right away, another 15 or 20 minutes. So it will probably be near 11:30 before the next time to break.

So either right now would be the natural point to break or approximately 11:30.

DR. CARBON: Let's take a break now.

(Recess.)

DR. CARBON: Let's go ahead, Mr. Sheridan.

(Slide.)

MR. SHERIDAN: My name is Bill Sheridan, Project Manager from Stone & Webster Engineering Corporation.

I will address plant design in the overview sense for the modular HTGR.

(Slide.)

The outline I will follow to indicate some of the key design selections that we have made as a result of going through the functional analysis, the integrated approach that Andy has previously described, then to take a look at our current plot plan, including the nuclear island and the energy conversion area, and then look in more focus in the nuclear island on the cutaway view of the reactor in the reactor building, in its position on the site.

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Then looking at the energy conversion area in terms of a typical -- or the flow diagram, which is representative of a typical HTGR plant, rather than looking at the hardware. But look at the flow diagram and the different connections between systems in that sense.

That is briefly the outline.

DR. MARK: Will you tell us what you think the relative costs of the nuclear island might be compared to the energy conversion facility?

MR. SHERIDAN: Can I come back to that?

DR. MARK: Yes. I am just asking will you tell us that later?

MR. SHERIDAN: I will present some information related to that.

(Slide.)

Overall plant key design selections, pursuant to and in compliance with the integrated approach that we have discussed previously.

First of all, we have a four-module plant with 350 megawatts thermal each module. So there's four modules arranged tandemly producing approximately 600 megawatts output, 558 megawatts electric.

DR. CARBON: This is presumably based on economics?

MR. SHERIDAN: This is based on economics. It is

based upon operations, goal one. It is based upon goal two. Goal zero of course is economics. And then it does meet the safety goals.

DR. CARBON: And the fact that you have four modular units rather than three or five or something is economics?

MR. SHERIDAN: Yes. Why four?

One, it represents that which the utility user perceives as his type of preferred configuration, that load level or that level of output. The 4- to 600-megawatt electric range is right now the preferred range.

One of the flexibilities with this modular setup and in the construction of it, you could operate one or two or three or four. So you have that flexibility.

This encompasses, this envelopes that consideration.

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The next significant selection made was two 300 megawatt electric turbine generators that provide the load for the four modules.

The next key selection is a single control room, which controls virtually automatically all of the modules in the two turbine generators.

DR. CARBON: We did understand correctly you could run one turbine on one module at 150 megawatts electric?

MR. SHERIDAN: That's correct. We can do that. And I think we'll see that when we get into the flow diagram of the energy conversion area.

The last key selection is the separation of the nuclear island, which contains the modules and associated equipment and the energy conversion area which contains the turbine and its associated equipment.

So there's a distinct separation between the nuclear island and the conversion area.

Looking at the next viewgraph, you have a foldout in the handout which shows the complete plot plan. What I'd like to do is to build that plot plan up in stages.

(Slide.)

Starting with the outline of the yard itself, the site itself, it's divided into two separate segments, the energy conversion area and the nuclear island. The nuclear

island is bound by the double security fence completely around it.

The energy conversion area has a normal type of fence boundary for that type of facility. The dimensions, we're talking approximately 900 feet in the north-south direction; the nuclear island east-west direction is approximately 600 feet; and another 900 feet in the east-west direction for the energy conversion area.

Those are approximate dimensions that may be of interest. The total land area less the switchyard is something on the order of 26 acres. The total land area out to the exclusion area boundary of 425 meters is approximately 140 acres.

So 26 acres for the site proper, 140 acres out to the site boundary, dimensions roughly 1,500 feet by 900 feet, including the switchyard.

Now we build up the plot. First, is the heart of each area outlined in red. The heart of the nuclear island of course is the core modules arranged in a north-south configuration. At the head of the tandom configuration is the reactor service building.

Now, the number isn't on here, but it's on your handout. That is item 4 on the plot plan that you have. Over in the energy conversion area is the turbine building.

DR. SIESS: Excuse me. Before you leave that

area, what I had on my handout shows a standby power building, which I assume is diesels.

MR. SHERIDAN: Yes.

DR. SIESS: There's a notable lack of separation there.

MR. SHERIDAN: Item 9 is the standby power building.

DR. SIESS: I assume there's two diesels?

MR. SHERIDAN: At the present time, it is contemplated to have two generators.

DR. SIESS: And they're side by side?

MR. SHERIDAN: Approximately side by side.

DR. SIESS: Have you considered sabotage in this plant design and layout?

MR. SHERIDAN: Sabotage is a feature that derives from I think virtually every one of the goals that we have. There are some specific features of the entire plant that enhance its sabotage resistance.

First, if you look at the dimensions of the sensitive area, the nuclear island area, it has been shrunken down. It's not the entire site. It's just one section of the site.

So your sensitive area has been minimized. Item 2, each one of these reactor modules is embedded in the ground such that the top of the reactor building is at

grade level. And we'll see that in the cutaway as we go on. That is an enhancement against security risks and sabotage.

Another way of looking at this is, within the island itself, you have dispursal. These four reactors are dispursed within the island. So that the concept of disbursal is an enhancement against sabotage.

DR. SIESS: That was just my point. It doesn't seem to apply to your emergency power.

MR. KELLEY: I think you should clarify that we have not identified a need for safety-related emergency power systems, such as diesels or gas turbines, at this time.

DR. SIESS: You don't need them?

MR. KELLEY: Right now, we don't believe that these would be safety-related. These would be here for availability and investment protection reasons.

DR. SIESS: Public health and safety doesn't require them?

MR. KELLEY: Public health and safety, we do not believe require diesels.

DR. SIESS: Nor does your 10 to the minus 5 on serious damage?

MR. KELLEY: From an investment protection viewpoint, we may need that to protect some of the equipment

in the plant. But, from a public health and safety view in terms of meeting those top level regulatory criteria, we believe they're not necessary.

DR. SIESS: Okay.

MR. SHERIDAN: That was my next point, Archie.

Moving on, rounding out some of the more important buildings on the nuclear island, we have number 5, the control building. Adjacent is the personnel services building and the waste management building, the rad waste building. And here we show item 9, the standby generators, which, as Archie said, are not safety-related based upon the functions and requirements.

That's not to say we don't have safety-related loads, but they would be handled separately through an uninterruptable power supply by batteries.

We do recognize there are class one type loads.

DR. SIESS: I assume the batteries are separated.

MR. SHERIDAN: The batteries are separated and the batteries are in the reactor service buildings, which would be -- we'll see them on the next escalation of the buildup, here, here, here and here.

This completes the final buildup of the plot plan. We have inserted items 2 and 3, auxilliary buildings, reactor auxilliary buildings. One set for the first two reactors and then a set down here, east and west reactor

auxilliary buildings for the third and fourth reactors.

Item 14, helium storage, pumping. Item 10, fuel for the standby generators. And item 8 is air-blast heat exchangers. We do have separate warehousing for the energy conversion area and the nuclear island.

Thirty-two is for the nuclear island; 33 for the energy conversion area. The standard buildings comprise the remainder of the energy conversion area; 25 the switchyard; 30 the fire pump house and water storage tanks; the configuration of 15, 16 and 17 is the auxilliary boiler maintenance building and water treatment building.

Item 21, nonessential switch gear. That about rounds out the totality of the plot plan. Now, what we would like to do is take a look at one of these reactor buildings. We'll be looking from the south to the north in elevation view. That will be the next viewgraph.

Looking from the south to the north...

DR. MARK: I'm interested that you mentioned south to north. It doesn't really matter, I suppose. I could put it in Arizona and have it go east-west instead.

MR. SHERIDAN: Yes. The only reason I did that is when you see this, you'll see the steam generator on the left side of the reactor vessel; and, before, you had seen it on the right side of the reactor vessel.

That's because we're looking from south to north.

It's just a matter of orientation.

(Slide.)

This is really the heart of the nuclear island. Tony Neylan will get into this in more detail. But I would just like to outline, one, the reactor building which is described -- by the way, this is a buildup in your viewgraph. You have a top enclosure on this; I'll put that up in a minute.

But the first thing to notice is the reactor building itself, which is outlined in concrete here. That comprises the reactor building, and that houses the steam generator and the reactor vessel.

The reactor building is set at grade level. This is grade level right here. The depth is 151 feet, approximately 151 feet. It's a right circular cylinder on down into the depth, and it has a diameter of approximately 63 feet.

The reactor building is a concrete building, approximately three feet thick here, two feet thick up there, and it's designed for seismic. Eighty-five percent of the site is designed for tornado and designed for missile protection.

DR. CARBON: For curiosity, what is the seismic?

MR. SHERIDAN: Point 3 and .15. That comes right from the utility user's guide. It's specified in there.

DR. MARK: What was the reference to missiles? I don't see any missiles coming in at the 151 foot cap.

MR. SHERIDAN: Right here is grade level. You have something above gr e. And that's one of the features of being embedded, is that the missile protection function is accomplished more easily. It's facilitated. That's one of the advantages.

DR. SIESS: Do you or the licensing staff see any problems with determining the seismic design for an embedded structure?

MR. SHERIDAN: At the present time?

DR. SIESS: Or a seismic analysis, shall I say?

MR. SHERIDAN: I'm going to ask Tony Sweeney from Bechtel to address that. They are doing the seismic analysis.

DR. SIESS: I'm sure they're doing it but I don't believe the staff has licensed an embedded structure in recent history. Have they explored with the staff what kind of questions on soil structure interaction?

MR. SWEENEY: If I understand the question correctly, yes. We are exploring all the likely questions that go into the soil structure interaction, and we're doing an analysis to look at that.

We understand it is an area where I guess we have to develop an understanding with NRC about how this will be

handled. So we're sort of using the state of the art, you know. The evolving ways of addressing this and the computer codes to develop that area.

DR. SIESS: And you are working with the staff people on that?

MR. SWEENEY: In the future, we will. I can't that, at this point, we have given them our analyses but our analyses are in progress. And, at the appropriate time, when we have completed our position, then we will interact with them.

DR. MARK: Suppose it's rather soggy where the site is chosen for water leaking into the minus 150 foot level?

MR. SWEENEY: If I might address that, first of all, we have a criterion, a user requirement, to be able to site the plant on 85 percent of sites that are available.

So, first of all, there are some sites that aren't acceptable for any nuclear and may not as well be acceptable for this plant.

We do believe that our water ingress control will sufficiently control water ingress and that shouldn't be a problem for the structure.

DR. SIESS: I don't see a sump.

MR. SWEENEY: There will be. All the things you would expect to be in an embedded structure will be there --

for water control.

MR. SHERIDAN: Some other dimensions of interest. The steam generator, some 85 feet tall by 15 feet on top, is the circulator.

DR. SIESS: Is that helical tube?

MR. SHERIDAN: Helical tube, yes. The reactor vessel is approximately 75 feet by 22 feet. Tony Neylan will go into that in more detail.

Now, I forgot. To complete the building...

(Slide.)

...the top of the reactor building, we have the maintenance bay building. This is a steel structure that is on top of the reactors and it goes completely from north to south, south to north. It encompasses all of that stretch of area covered by the four reactors.

DR. SIESS: How much of what's on the slide is safety-related?

MR. SHERIDAN: How much on this total viewgraph is safety-related? I'm going to let Fred Silady address that question. Fred...

DR. SIESS: It might be easier to say how much is not safety-related.

MR. SILADY: Probably, the best way to approach the question would be to go through the licensing approach and talk about the process by which we term things

safety-related and then, at the end of that, tell you the status.

DR. SIESS: Assume that I know that, because I do.

MR. SILADY: Roughly, the fuel, the vessel, the graphite core.

DR. SIESS: Everything on the righthand side here?

MR. SILADY: I wouldn't go so far as to say everything on the righthand side. And even on those things that I'm mentioning, it is a particular function that will be safety-related.

For example, on the vessel, it will be a function to control chemical attack. It may not be a function to retain primary coolant. So, as we go through the presentation that's coming up here in a few minutes, we'll get into a little more depth on that.

DR. SIESS: I'll wait.

MR. SHERIDAN: But in the maintenance bay building is a 150-ton bridge crane. It runs up and down from the reactor sites. In the power conversion area, the turbine building is a standard turbine building, approximately 200 feet by 200 feet.

I think it's of interest to look at the turbine cycle in terms of a simplified flow diagram.

(Slide.)

There are four steam generators producing the steam at generally fossil-type of steam conditions. That is, 1,000 degrees Fahrenheit, 2,500 pounds pressure. That's the output of the four steam generators. They are headed together but isolable.

Normal operation would have them headered. Off that header come the two turbine generators with the HP/IP. That's a little on the spot correction. That should be IP instead of LP in that middle bank of turbines there. Intermediate pressure, and then a low pressure turbine to the generator, exhausting at 2.5 inches of mercury to the condenser. The condensate pump feedwater heaters. The de-aerating feed tank feedpumps on into a feed header, which is combined.

In other words, the feed is headered and the steam is headered. It will have the capability of isolation of any of the steam generators and ability to provide steam from a steam generator to a turbine.

So there will be a valving arrangement to provide that type of flexibility. That's the significant part I think of this viewgraph, is the headering of the steam and the feed. The rest is rather standard power plant.

(Slide.)

In closing the overall plant parameters, the

entire plant, some of the larger parameters, significant parameters, are core power, which is of course four times 350 to 1,400 megawatts thermal.

The gross power to the turbine is slightly above that due to the circulator heat input and minus the ambient losses, and some other losses that we can get into. Steam pressure, conventional; fossil parameters so to speak, 1,000 degrees Fahrenheit; 2,415 psia exhaust, rather standard exhaust pressure; two and a half inches of mercury, total generation 600; auxilliary power requires some 42 megawatts, with a net generation of 558 and an overall plant efficiency of approximately 40 percent.

And that's the total plant layout and some of the important parameters.

DR. MARK: Now, the numbers on that slide, I believe you have said were chosen as the result of a survey of the utility representatives. That is, this is about a good size. This modular feature is good.

Is that mainly influenced by U.S. consumers, utilities? Or is it also with a view to outside the country interests?

MR. SHERIDAN: The requirements for the particular thing?

DR. MARK: 600 megawatts.

MR. SHERIDAN: That is from U.S. user utility requirements. It so happens that this plant does have the feature of relying on a single monitor or two monitors which could be sold elsewhere, but the totality of these parameters really derives from U.S. user requirements.

DR. MARK: Now also the question you showed us in one of the early slides, you're securing area, the extra barbed wire and stuff, enclosed something, but didn't enclose everything. I was wondering why are the diesel geneators outside inside of inside that barbed wire?

MR. SHERIDAN: Let me put the Vugraph back on.

The barbed wire doesn't really show up very well.

DR. MARK: No, I hope not.

DR. CARBON: They're inside, Carson.

MR. SHERIDAN: What I said is, the barbed wire markings don't show up very well. It's difficult to see them.

DR. MARK: Where are the diesels then?

MR. SHERIDAN: The standby generators are located here, you see in outline form.

DR. MARK: So my wild-eyed saboteur can't get at

them.

MR. SHERIDAN: Let's make it clear --

DR. CARBON: I think it's clear now, and maybe we'd better move ahead.

DR. MARK: Now you were going to tell me about the relative cost.

MR. SHERIDAN: I was going to tell you about the relative cost. The total cost of the plant is about $1 billion, not including AFDUC. I think Archie Kelley has some figures on the division of them. We are working on the division between the nuclear island and energy conversion areas. I think Archie has some of those figures, but that is a feature in our cost estimate. We want to be able to segregate those things and be able to push a button and say, here's what this is on the nuclear island, so we're working towards that.

Archie, do we have any initial figures on that yet?

MR. KELLEY: The split is roughly 700 nuclear island, 300, including development costs.

MR. SHERIDAN: 700 nuclear island, 300 energy conversion, including development costs.

DR. MARK: I was asking about it, because that 300 is pretty independent as to whether it w s nu lear or nonnuclear.

MR. SHERIDAN: Yes.

DR. MARK: Thank you.

MR. SHERIDAN: Going back to the road map, Tony Neylan will now discuss the nuclear island and really focus in on the NSSS, and we are here on the road map discussing plant design.

(Slide.)

MR. NEYLAN: Good morning, gentlemen. I'm Tony Neylan with GA. I am a division director responsible for the HTGR programs under the DOE contracts, and I'm going to give you a brief overviw. The handouts that you have are in a slightly different order. I will be trying to catch up a little bit of time by not completely reading through of the text ones that are on there.

(Slide.)

I will also be coming back later after the licensing approach to talk about the nuclear steam supply in more detail. Specifically, I will be at that time addressing each one of the nuclear steam supply systems that are identified on this Vugraph, which identifies those systems which comprise the nuclear steam supply scope with the exception, in fact, of the reactor cavity cooling system, which was on the last presentation, in fact, as part of the nuclear island in the balance of plant.

(Slide.)

In this overview, I will concentrate on the configuration and design parameters, and first of all, which is slightly out of order in your handout --

(Slide.)

-- addressing the design parameters. And this is one module. I think that's about four sheets in there. One module of the overall heat balance diagram that Bill Sheridan showed you, and it indicates on here a single reactor core generating 350 megawatts thermal, when supplied with helium at approximately 500 degrees and delivering an average temperature of 1268 degrees F from the core. The nominal pressure of the primary circuit is 1925 psia.

The steam generator will deliver 352 megawatts thermal when supplied with cooling water at 380 degrees F, 3000 psi, and supply steam at 1005, 2500 psia. These numbers should be consistent with the numbers shown on the larger diagram. Of course, this then goes into the header system and the multiplier reactor modules to the twin turbines. The helium is pumped through the primary circuit with a helium circulator, which develops a total head of about 13.2 psid. Those figures are, in fact, the key ones.

(Slide.)

They are shown on this next sheet in there. I won't read them thorugh for you again. You may note that, in fact, the outlet temperature of this HTGR is less than

Fort St. Vrain, which delivers about 1400 degrees F average core outlet temperature.

DR. MARK: Could you make just a rough comment on a couple of vague questions?

Supposing we found ourselves using plants of this sort. The helium requirements and the helium supply, how do those stand out?

Also you mentioned a DOE contract. What's the rough size of that contract?

MR. NEYLAN: The contract from DOE is part of the HTGR national budget which averages around about $30 million per year. The contract to GA is approximately $15 million.

The helium supply is a completely contained system, and therefore, there is an initial fill of helium, and then a small makeup, which is expected to be less than 1 percent of the inventory per year, including all helium leakages, and so forth.

DR. MARK: But our national supply of helium is finite and has even been reduced since 10 years ago.

MR. NEYLAN: Based on the projected market 10 years ago, there was a lot larger plant with a lot greater requirement for helium. There is perceived to be no helium inventory supply problem, and there still appear not to be, even with the large economy of these kind of reactors.

DR. MARK: Thank you.

(Slide.)

MR. NEYLAN: I'd now like to talk about the configuration. You saw this diagram in a reactor structure that was embedded below grade. The features of this are that the reactor core is in a separate reactor vessel from the steam generator. The reactor vessel and the steam generator vessel represent existing lightwater technology. They use the same materials and are approximately the same size and somewhat lesser weight than actual vessels which have been supplied for the PWR industry.

The diameter of the reactor vessel is 24 feet -- 22 feet, pardon me. The height is about 72 feet and the control rods stand out a further six feet. The steam generator vessel is approximately 14 feet in diameter and 83 feet high. They are connected by a coaxial cross duct of approximately six feet in diameter.

DR. SIESS: That vessel's about the size of a BWR vessel??

MR. NEYLAN: I will this afternoon show you a diagram which puts them on one sheet and compares them for you.

DR. MARK: Questions that we are often plagued stress corrosion cracking, do not apply to this vessel at all, I suppose.

MR. NEYLAN: In terms of the water environment

versus helium, that is true.

DR. MARK: But when you get across to the steam generator, then you've got the old stuff back.

MR. NEYLAN: I will again this afternoon show you some details of that. It is a helically coiled steam generator with water inside the tubes, and the outside of the tubes is a helium environment. As you'll see, in fact, on the next schematic.

I'll go through the flow diagram and show you where the helium is.

The steam generator vessel is located below the level of the reactor core, which in subsequent potential pressured heat-up events or cool-down events, will preclude potential natural convection into this vessel and thereby protect the steam generator. The main circulator is located in the cold leg at the top of the steam generator. It is a submerged electric drive motor.

Located below here is an independent shutdown heat exchanger with its own circulator. I will come up to each one of those and describe them in a little more detail. Those statements --

DR. SIESS: Tony, you'll talk about the bearing lubrication system?

MR. NEYLAN: This afternoon.

DR. CARBON: Is the Fort St. Vrain steam

generator down or up?

MR. NEYLAN: The Fort St. Vrain steam generator is in a single cavity vessel located beneath the core.

(Slide.)

The statements that I just made about the locations and so forth are on another sheet in your handout. I won't spend time on that.

(Slide.)

The next Vugraph traces the helium flow during normal operation, and starting with the cold helium from the circulator, it flows on the outside of the coaxial duct. It flows into the outer areas of the reactor vessel and, in fact, is channeled through to the top of the core. It flows down through the core, as in Fort St. Vrain, across the hot duct, into the steam generator, where it flows down from the steam generator.

The steam generator is an uphill boiling, once through, steam generator, around and back through to the circulator. Feedwater is in at the bottom, helically wound once through and then out on the side.

If we can focus in for a moment on the core.

(Slide.)

The fuel components rely on the basic fuel particle technology that has been developed and demonstrated on Fort St. Vrain and is evolving on the low enriched

fuels, at this point, to further develop this technology.

The basic fissile and fertile particles. The fissile particle is U 235. You can ignore this, because you can't have an opportunity to recycle it, if you ever wanted to.

In this reactor, the current fuel cycle is once through, then out into permanent storage. It's mixed with the fertile particle into a fuel rod like at Fort St. Vrain, that's approximately a quarter inch in diameter, two inches high. These fuel elements are located in fuel holes in a graphite block next to coolant holes. The coolant flow is down through the block. These block are approximately 14 in hes across, flat, 32 inches high. They are the same as Fort St. Vrain. They're stacked one on top of another in a hexagonal arrangements.

DR. MARK: You mentioned U 235 and U 233. Is plutonium not an alternate possibility?

MR. NEYLAN: As a plutonium burner, it is, but it is not part of this graphite design.

DR. MARK: Why is it excluded? Just because of publicity, or what?

MR. NEYLAN: It's basically, a low enriched uranium cycle has been selected to meet with proliferation requirements and to meet with the user requirements.

DR. MARK: Publicity. Proliferation. It's not

because of mechanical or nuclear.

MR. NEYLAN: That's correct. The same reactor has the capability with the same plant design to work on fully enriched cores like Fort St. Vrain, or in fact, on recycled cores with the 233 or even with plutonium.

DR. MARK: You say fertile thorium. You could just as well have fertile uranium.

MR. NEYLAN: Yes.

DR. SIESS: And you do intend to have the fertile thorium in there?

MR. NEYLAN: Yes.

DR. SIESS: Just as a contingency of recycling?

MR. NEYLAN: Not only for that. It does develop a two-particle system which could be used for recycle at some future date. It is used for fuel management within the core, in the design of the core.

DR. MARK: Now thorium 232 is converted to uranium 233 for recycle. It leaves you with a very messy radiation problem in connection with the fuel propagation, as compared to going from U 238 to plutonium 239. The gamma rays from uranium 233, I guess.

MR. NEYLAN: We are not proposing recycle as part of this. This is a low enriched. We do not intend to recycle the generated U 233.

DR. SIESS: Do you get energy from it?

MR. NEYLAN: Does anyone have an answer to that?
MR. SILADY: Yes.
DR. SIESS: A significant amount?
MR. NEYLAN: I don't think so.
DR. MARK: Low enriched? How low?
MR. NEYLAN: 19.9 percent.

DR. MARK: Would you be better off with 20.3 percent?

MR. NEYLAN: From a regulatory point of view, no. That gets into safeguards issues.

From an economics point of view, 97 percent would be better fully enriched, as on Fort St. Vrain.

(Slide.)

An important feature of the design is the configuration of the reactor core. This viewgraph shows a planned view of the core.

The hexagonal blocks that I showed on the previous diagram are arranged in an annular region of essentially three rings of fuel elements, which are patched in this diagram.

The central region are graphite blocks of a similar size but are not fuel.

The outer region also has two rows of fuel blocks, removable reflector, and there are permanent reflectors outside of that, all supported within a core barrel from the steel vessel.

The selection, as will be apparent later in the day, is important in governing the temperatures and the fission product releases that result under various cooldown and excursion events.

The control rods are located in the outer

1 DAVbur

reflector. They are 24 in number. There are six located in the inner graphite columns, and there are independent reserve shutdown channels, 12 in number, indicated by the black dots on the diagram.

DR. MARK: This is uranium oxide or uranium metal?

MR. NEYLAN: Uranium oxide, oxycarbide.

DR. MARK: Supposing the regulatory gurus said, okay, you car use 30 percent uranium if you want, would this diagram change very much?

MR. NEYLAN: The diagram per se would likely not change at all. The loading -- the fuel loading in an individual block might change, though.

DR. MARK: So you could really use that diagram with a little shifting all the way up to 90 percent enrichment?

MR. NEYLAN: That is correct.

DR. CARBON: What is the reserve shutdown system?

MR. NEYLAN: The reserve shutdown system are pellets of boronated graphite like those on Fort St. Vrain, which are released from hoppers and go into holes in the fuel blocks.

DR. CARBON: And what are the boronated pins there?

MR. NEYLAN: They are provided for shielding to

reduce the doses to the steel vessel.

DR. CARBON: Are they all the way around?

MR. NEYLAN: Yes, they are. That is diagrammatically shown.

(Slide.)

I thought you would be interested in the ways, in this overview, at least of removing the heat from the core. The normal heat, of course, is removed by the heat transport system. Decay heat is also removed by the heat transport system in the same flow diagram that I showed you previously.

We have two other means of removing decay heat from the core. One is termed the shutdown cooling systems, and on each one of these I will come back here later this afternoon and discuss it in more detail.

(Slide.)

In the shutdown cooling system which is provided here, essentially for operations and maintenance reasons in the event that the circulator or normal steam generator is not available, which allows you to do maintenance operations more quickly than if you had to wait for the whole system to cool down naturally.

Its flow is from the circulator in the outside of the vessel down through the core. Then instead of going across into the steam generator vessel, it goes down through

a small heat exchanger, down into the circulator, and back through this system.

So it is here a small maintenance loop provided in the facility.

(Slide.)

The ultimate means of removing decay heat when all primary coolant flow is not available to you -- in other words, you have lost the heat transfer system, you have lost the shutdown cooling system -- is by conduction and radiation through the vessel to a passive air cooling system located in the cavity, which was indicated on the general arrangement drawings that Bill Sheridan showed.

And again, I will come back later this afternoon and show you some details on that.

(Slide.)

I did, however, want to point out while I am here the resulting temperatures within the primary system and specifically the peak temperatures within the core that result from this radiation conduction mode to the reactor cavity cooling system, the RCCS.

And you can see that the peak temperatures are somewhat in excess of 1600 degrees C. Some 30 or so hours, 50 hours into the event, there will be a gradually cooling down.

Again, Fred Silady will provide a little more

detail on this. But I wanted to put that in perspective with the resulting effect on the fuel particles.

(Slide.)

This is data based on the triso-coated UCO particles which were developed a number of years ago, and there is continuing development in this area, but it indicates that the important point is that the fission gas releases with temperature are very low and are expressed here as a failure fraction of the fuel.

But clearly, if one is able to maintain temperatures in this kind of range, then the resulting fission gas release which contributes to accident releases are very small. That is a key in the inherent safety of this system.

DR. MARK: What is the melting temperature of uranium oxide?

MR. NEYLAN: The silicon carbide starts to break down in this kind of temperature range, 2200. That is the breakdown of the fuel particle. That is the primary barrier to release of the fission products.

DR. MARK: That is the most temperature sensitive element in that core configuration?

MR. NEYLAN: That is correct.

DR. MARK: Steel?

MR. NEYLAN: There is no steel in the core at

this time.

We will show you the resulting temperatures on the other core supports, structural components and the vessel, and they are all within acceptable limits.

DR. MARK: Graphite is really quite a bit higher, anyway?

MR. NEYLAN: That is correct. In fact, it is still increasing in strength at this point.

The key, then, to this design is the selection of the fuel, its performance, the configuration of the core, which allows you to limit this temperature and hence fission product release, and the configuration -- the cooling configuration which allows you to get rid of -- dissipate the decay heat generated in an acceptable and passive way, based on that conceptual design that I have just made.

We believe that the inherent characteristics and the passive safety features of this modular HTGR will fully meet those quantified goals and objectives of economics, investment risk aversion, and safety.

The key ones to remind you of are the 10 percent economic advantage versus coal, the 10 percent unavailability from an economic and operational point of view, the investment risk, the loss of plant at 10 to the minus 5, and the PAG sheltering guidelines, the 5 rem thyroid, 1 rem whole body, and 5 times 10 to the minus 7.

With that, I would like to end this overview and come back and give you some of the details this afternoon.

DR. MARK: Just before that maybe, Fort St. Vrain has been plagued with moisture.

What have you done about that?

MR. NEYLAN: It has, indeed. That was relevant to Dr. Siess' question.

We are very concerned about that, and we are looking at the sources and have been actively looking at the sources, the lessons learned on Fort St. Vrain to correct that problem.

(Slide.)

DR. CARBON: Mr. Silady, anything you can do to gain us some time would be very helpful. We are 50 minutes behind.

MR. SILADY: All right.

My name is Fred Silady, Manager of Safety and Reliability at GA.

The topic of my presentation this morning will be to continue on this master diagram and proceed from the process, the design approach that we have taken, to develop licensing bases that are specific to the HTGR and apply the process to come up with things such as licensing basis events.

My presentation is in two parts. There is a very

brief overview that goes into, in a capsule way, the development of the licensing bases.

(Slide.)

Let me just remind you of our licensing approach that was touched on briefly this morning. There are three steps.

It is tied to the top level regulatory criteria as a starting point.

The second is the area called the bridge, to develop a process to derive licensing bases specific to the modular HTGR.

The third is the application of that bridge.

We have heard this discussion this morning. I am going to give you a brief capsule of the process. Then I am going to go into a longer presentation that goes into more depth in terms of the process and its application step by step.

(Slide.)

First, a brief overview.

We are going to identify the likelihood of events and classify the events into three regions for a comparison against the top level regulatory criteria. This classification of events, using risk assessment, will form the basis then for the licensing basis events.

We are going to examine those events to identify

the functions that we are going to rely upon in order to meet the top level regulatory criteria. These functions that we rely on are the basis for our principal design criteria.

The third step is to choose design selections to accomplish those functions that we are relying upon to meet the top level regulatory criteria. This step leads directly into which systems, structures, and components. Design selections are those which should be termed safety related to corresponding quality assurance.

DR. MARK: What do you think of when you say top level regulatory criteria? Do you mean Part 100?

MR. SILADY: It is as we discussed earlier in Kelley's presentation. It is those quantitative doses or risks to the public health or environment that are plant independent, that are top level, that are quantifiable.

With that brief overview, let me go into much detail.

(Slide.)

The presentation outline is arranged in a fashion where each of the three areas, the three identified licensing bases, is discussed in terms of first defining some terms, then looking at the step-by-step method with an example application as I go through so that you get a better handle on exactly what the method is.

First, I am going to go through the licensing basis events. Then I am going to talk about the functions that we rely upon in order to meet the top level regulatory criteria in 10 CFR 100, then the equipment classification in the same manner, and then a summary.

(Slide.)

Starting in with the licensing basis events, these are those off-normal or accident events used for demonstrating the design compliance with the regulatory criteria.

Collectively, the licensing basis events are analyzed in risk assessments for demonstrating compliance with the interim safety risk goals.

We have defined three kinds of licensing basis events, which I will now define one by one.

(Slide.)

The first of these we call anticipated operational occurrences. These are events that are expected once or more in the plant lifetime. Therefore, these are events whose frequency we would expect to be greater than once in 40 years, or .025 per year.

We analyze these events realistically, and they typically appear in Chapter 11 of SARs, and they are compared against 10 CFR 50, Appendix I.

So the anticipated operational occurrences are

compared against Appendix I, which was one of our top level regulatory criteria.

(Slide.)

Design basis events we define as events of lower frequency than the anticipated operational occurrences; that is, they are not expected to occur in the lifetime of one plant. However, we do expect them to occur in the lifetime of a large number of plants, say 100 plants.

So typically, we think of these events as falling within the range of once in 40 years down to approximately once in 10,000 years, or a 10 to the minus 4th frequency. If we had 100 plants, one would expect a design basis event somewhere in the lifetime of all 100 of those plants.

The consequences of design basis events are analyzed conservatively in Chapter 15 of the PSAR, now against 10 CFR 100, another one of our top level regulatory criteria.

(Slide.)

The third of the three licensing basis events we call emergency planning basis events. These events are still lower frequency than the design basis events. They are not expected to occur even in the lifetime of all standard HTGRs, where now we are taking this to be 100 plants. So they have frequency below 10 to the minus 4th.

However, we do have a lower frequency for the

emergency planning basis events, and we take this to be 5 times 10 to the minus 7th per reactor year for consistency with the interim NRC risk goals, particularly the acute individual risk goal that is specified at 5 times 10 to the minus 7th.

The consequences of these are realistically analyzed in risk assessments and in Chapter 13. These emergency planning basis events would be shown to comply with the projected action guideline dose limits at the emergency planning zone.

DR. SIESS: Is there any consistency between this and the PAG dose limits of the previous PAG's of part 100? I can't quite picture how you could get to this stage and meet the PAG and not automatically meet part 100 of the DBE's. These events are worse than DBE's and the doses are a fraction of the DBE's.

MR. SILADY: The events are lower in frequency. They may or may not have consequences greater than the higher infrequency design basis events. The basis on which emergency plan basis events are evaluated is the emergency planning zone.

Our user requirement is that that distance be at the exclusionary boundary. But, in terms of getting a license for an HTGR, that distance could vary and could be at five miles, like it is at Fort St. Vrain, or 10 miles for existing reactors.

DR. SIESS: I'm still confused. If I've got a 10 to the minus 4 or 10 to the minus 5 event, because 25 rem whole body --

MR. SILADY: I have a diagram that may clear this up.

DR. SIESS: I'll wait.

DR. MARK: In talking about Part 100 PAG's and stuff, all of this is without the use of containment.

MR. SILADY: At this point, I'm describing an

approach that is generic, an approach that's going to take the blue part of our master diagram -- that is, the design that we have evolved from the rigorous process that you've heard about, and translate over into licensing bases specific to the modular HTGR.

So, yes, specific application of this process may indeed lead to meeting the 10 CFR 100 at the DAB and meeting the regulatory actions at the emergency planning zone without the use of a standard light water reactor containment.

The process, however, could be applied to any nuclear power plant. Let me go to the first of many steps, and I hope to be able to come back to Dr. Siess' point to clear up any confusion with regard to the relationship of 10 CFR 100 and design basis events to the protective action guidelines and emergency planning basis events.

(Slide.)

The first two steps in the process are to define three regions on a frequency consequence plot, risk plot. The second step is to compare the risk assessment of the design to that frequency consequence risk plot.

Subsequent steps then will take each of those classes of events and choose from those, select from those, ones that would be anticipated operational occurrences design basis events, or emergency planning basis events.

The subsequent steps will then also indicate how each of those are evaluated and against which of the top level regulatory criteria.

But, first, let me show you a typical risk plot...

(Slide.)

...with the regions shown according to the definitions we've just gone through.

The first region is the anticipated operational occurrence region, where the ordinate is frequency and infrequency per plant year. The abscissa is mean whole body gamma dose at the exclusion area boundary. The slanted line is appendix I. On the annualized basis of five millirem per year.

It extends down to once in a lifetime of one plant, which is once in 40 years.

The second region we have defined is the design basis region. It starts from that point and extends down to 10 to the minus 4th, which corresponds to events that would be expected to occur in the lifetime of, say, 100 plants but not expected in the lifetime of one plant.

The third region starts from that point and goes beyond these design basis events into the emergency planning basis region. It extends down to 5 times 10 to the minus 7. In each of these regions there is a top level regulatory

criterion that the events are evaluated against. In this region, it's Appendix I. In this region, it's 10 CFR 100, 25 rem that you just referred to, where we have shown a slant in the line in recognition of the fact that, for higher frequency events within this region, one wouldn't design all the way up to 100 percent of Part 100.

So this is at 10 percent at this point. And then the third region, the protective action guidelines of one rem that we heard about earlier, isn't shown because this is the dose at the exclusion area boundary. It is our user requirement that all events in here and all events in here be below one rem, so that there need not be any offsite evacuation.

So, if I was plotting user requirements on here in addition to top level regulatory criteria, I would have a line coming straight down. Of course, we're going to have to have agreement if indeed we want to have our exclusion area boundary be our emergency planning zone that all our events are within that region.

But it could be that we could license a plant with an emergency planning zone much greater than the EAB. That's the practice of existing plants today, in which case, the dose at the EAB is much greater than that one rem. So the line isn't shown.

Did that help at all?

DR. SIESS: That helps but where would the safety goal put you if you knew what it was?

MR. SILADY: The safety goal is off the chart. In essence, since I'm only at 100 rem here, and particularly an acute fatality individual risk.

DR. SIESS: You take the LD-50 for acute, the dose that will kill half the people, half those exposed to it? Is it 400 rem, 250 rem?

MR. SILADY: Right. There's a function in there in terms of the probability of dying versus dose. And that would be considered.

DR. SIESS: And that would be acute fatalities?

MR. SILADY: Yes. And, in terms of latent cancers, similarly.

DR. SIESS: It's manrem then. In terms of latent cancers, you have to reintegrate, don't you?

MR. SILADY: That's correct. Later, the last step in the licensing basis event methodology, I will show you the results of our preliminary risk assessment summed over all the events, with the conversion of the consequences to latent cancers and compare that to the 2 times 10 to the minus 6 latent cancer individual risks and goals.

But I'm getting ahead of myself a little bit. I'm still just on step one, making sure that we have a good base here before proceeding, that this is a center or the

heart of our approach to the licensing basis events, is the definition of these regions.

DR. CARBON: How much agreement have you had with the staff on that?

MR. SILADY: As discussed earlier today, we have made presentations to them on this approach. We have made the documentation in a form that's being readied within the next month to be sent to them.

(Slide.)

The second step was to compare a risk assessment to these three regions on the risk plot. These are early results for the 350 plant, and not all the results are plotted, just those that are dominant of the ones that we have examined at this point.

And my purpose in showing this is not to go through and define each of these release categories in the event of...but again to show you conceptually the approach. There will be events throughout all three regions. Some will fall into the AOO region, others into the design basis region, and others of lower frequency and perhaps higher consequences, perhaps not, in the emergency planning basis region.

Shown on the chart is the mean value in terms of frequency. And the mean value in terms of consequences with uncertainty bands. The uncertainty bands play an important

role.

DR. SIESS: Another thing that I have to keep in mind when I look at that is that, almost without exception, when the staff reviews somebody else's PRA, the results tend to move upward by a factor of 1 to 2 orders of magnitude.

MR. SILADY: Yes. We are aware of that. One thing I should have mentioned at the beginning of the presentation is that this process that I'm describing, of getting from the end nuclear power plant design product over to the licensing stage will be repeated at each stage of the design.

So our knowledge, our risk assessments at this point of a conceptual design, may change as we go into preliminary and final design.

We are hoping that the characteristics of this concept are such that we wouldn't anticipate a large difference in opinion between us and the staff.

This provides the basis for which we can have discussions in those areas.

DR. SIESS: Now if they only move up and don't move to the right, that doesn't really give anybody much of a problem, because these are all so far over.

MR. SILADY: That's an excellent point, particularly with this concept, which is more a low consequence concept rather than a low frequency concept.

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And if we can show that the safety is transparent with the passive features, we may be able to have less movement right to left in terms of consequences.

DR. SIESS: I think that's a key to much of this.

MR. SILADY: I agree.

(Slide.)

The next two steps in the process are to identify those families of events that fall within the AOO region and to identify that rely upon some design selection to keep them within that region, not to violate Appendix I, to identify those as AOO's. There are some typos on this chart.

Then to evaluate the consequences of those selected AOO's realistically against Appendix I. Let me show you an example of that on the next page.

(Slide.)

This is very conceptual but it takes one of our real points that we're seeing in our risk assessment so far, which happens to be a primary coolant leak releasing a small amount of the circulating activity to the environment.

The point is assessed at being within our goal. However, if it were not for specific design selections that we consciously made to control release, the point could be out here.

We intentionally have very good fuel quality specified. That makes the circulating activity quite low. So even the uncertainty band doesn't exceed the Appendix I region goal. But we would take this event and call it an anticipated operational occurrence and evaluate it and report on it in the PSID in Chapter 11.

DR. MARK: What is the main component of your circulating activity?

MR. SILADY: Primarily, Krypton 85, 88.

DR. MARK: It's gas from the fuel elements that might possibly escape, or might possibly get into the helium.

MR. SILADY: The circulating activity arises as a result of as manufactured defects in billions of little fuel particles, such that the coatings aren't properly coated, or a very small fraction.

And, in addition, there is during normal operation some statistical failure of a very small fraction of those. Those both can release gases into the circulating activity.

(Slide.)

The next three steps do an analagous thing in the design basis region. There's one extra step thrown in to account for uncertainties. We identify as design basis events those families of events within the design basis

region in much the same manner as we did for the anticipated operational occurrences.

However, recognizing that the risk assessments have uncertainties and those uncertainties may cause the point to be on the border of the defined regions, we say that if the upper or lower bound frequencies extend into the design basis region, then those too are candidates to be design basis events.

And, repeating, these events are evaluated conservatively and shown against 10 CFR 100 and Chapter 15. And there's a page, a couple of pages on gives us an example. I'm going to skip one of the pages in the package, since it comes up again.

(Slide.)

In light of the time. And I think we will adequately cover it when we get to it a little bit later.

So, here, we have this particular release category family of events and there is a design selection that keeps the point here rather than in an unacceptable region that exceeds 10 CFR 100.

So the function of controlling release is accomplished by some design selections. And I'll save that story just for a few more minutes, until we get into the second and third segments of this presentation.

(Slide.)

The emergency planning basis events. The two steps with regard to these, the final of the three kinds of licensing basis events are just to examine that region that is beyond the design basis. Find those events that are consequence dominant, the key ones, to turn those emergency planning basis events and to evaluate those realistically in terms of siting evaluations.

And I'm going to skip then the next page in the package, which shows one of those being selected, the one with the higher consequences.

(Slide.)

Step 10 and the final step then for the licensing basis events is to compare the risk assessment of the design that was developed with the integrated approach to the interim NRC safety risk goals. That's shown in the next table.

(Slide.)

The first column shows the regions for the events as they are assessed to lie in -- the anticipated operational occurrence region, the design basis region, the emergency planning basis region. These are the dominant events.

The list is quite a bit longer, of course. But one takes the frequency in events per plant year, takes those doses -- whether it be whole body, thyroid, or

whatever -- converts it to cancers per event.

These two mean values are multiplied together and the risks summed to a value at this point of 2 times 10 to the minus 8th latent cancers per year.

That's compared to the risk goal of 2 times 10 to the minus 6th. And at this stage, we have a margin of approximately 2 orders of magnitude.

Note that the majority of the risk is not coming from the design basis region, or even from that region beyond or the emergency planning basis region, but from the anticipated operational occurrences.

So, here, already, at the conceptual design, we are getting some insights about our design and its risk envelope.

The second of three segments of my presentation are the required functions, the functions that we rely upon in order to meet 10 CFR 100. And it builds on the selection of the licensing basis events.

(Slide.)

First, some definitions. These functions are those that are needed to limit the radionuclide retention to meet 10 CFR 100 doses for design basis events. These functions and how we accomplish these functions are the basis for the principal design criteria that we will be back later in the year talking to you about. And of course the

principal design criteria go in Chapter 3 in terms of showing how we comply with the top level regulatory criteria.

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MR. MILLUNZI: Fred, if I could on that first one, the word "meet," that means those functions which we are going to rely upon.

MR. SILADY: A very good point. One which I think will be brought out in the next slide or two.

There are many ways that one could mean 10 CFR 100, even for a specific design.

(Slide.)

There are two steps in this selection of functions that we're going to rely upon to meet 10 CFR 100. The first is to identify those functions required to meet the top level regulatory criteria and the user utility safety requirements.

So first, I will try to find all the functions tht have to do with radionuclide retention.

(Slide.)

An example of that is this segment out of our functional analysis tree. That is the Goal 3 portion of the tree, down to four levels. The top box reads "Maintain control of radionuclide release." And these two ways are the ways that the plant does that. Control personnel access and control radiation. In turn, there are three ways to control radiation. One of those is, control radiation from the core. Others are from processes and from storage. There are two things one needs to do, in order to control

radiation from the core. Control the direct radiation and control radiation transport. There are four ways, required functions to control radiation transport. You can control transport from the core, in the primary circuit, from the reactor building and from the site, expanding the core on down one more layer.

One can retain radionuclides in the fuel particles and retain radionuclides in the core graphite down to this level. These are all the functions that we need in order to accomplish all of the top level regulatory criteria and all of the user requirements relative to not needing evacuation, and so on. But that is just one of the process.

DR. SIESS: I asked the question earlier about defense in depth, and I'd like to point out that some people, defense in depth, are the various levels you've got here. There are at least three different definitions of defense in depth, but one is in terms of barriers. I think you've got more than four levels here, but I'm not sure.

MR. SILADY: One can think of it in terms of barriers, and that's the way the term has evolved in the industry to date, in terms of a quieting, a vessel, a containment and a site.

DR. SIESS: That's one impression of it. The other one is the one that Mr. Millunzi gave.

MR. SILADY: We believe that the one Mr. Millunzi discussed is the way it originally started off, the original definition, and that there are many ways to accomplish defense in depth and, indeed, to meet your top level regulatory criteria, you will see that we are utilizing each of these in meeting the combined top level regulatory criteria and user requirements. Our site is 425 meters. We have a reactor building. We have a primary coolant boundary, and we certainly have good fuel particles; however, the point I want to make in step 2 is, which of these are we relying upon, in order to meet 10 CFR 100.

(Slide.)

Select those radionuclide retention functions required to meet 10 CFR 100 does for design basis events.

And your next chart, if you're flipping ahead, shows the functions shaded that we need for 10 CFR 100.

(Slide.)

Based on our present plant design and assessment, we feel that we can show that the sources of radiation and radionuclide release from processes and from storage are quite small. Similarly, we feel that our sources within these other areas here are quite small, and I would give you some numbers on those in just a minute.

So in terms of meeting 10 CFR 100, the key thing and the only thing we need to do is retain the radionclides

in the fuel particles. And from here, I wish to go into a fuel of the design basis events and a few of their thermal and fission product release characteristics and then come back to another sphere of functions that we have found that we need, in order to keep the fission products within the fuel particles.

(Slide.)

Before proceeding, I do need to spend just a quick minute, although it's in your package, and I won't go into it in a lot of depth, to see that first we have to take what 10 CFR 100 is. That is 300 rem thyroid, as an example, or 150 rem at the construction permit and translate that into what are the allowable curies that could be released from the fuel. And we have done this, and we have made assumptions consistent with Reg Guide 1.4 to come up with 250 curies of iodine that cut be released from the fuel and without any other consideration of hold up in the vessel or in the reactor building, exactly match 150 rem.

For longer term releases, the weather and the breathing rate and the chi over Qs change and the number is higher. Keep those numbers in mind, 250 curies for short-term release and 900 for long-term.

DR. SIESS: Is there a frequency of probability thinking that's been assigned to the atmospheric dispersion? Is that the average?

MR. SILADY: It's directly from Reg Guide 1.4. It's the conservative chi over Qs.

(Slide.)

This page compaes some of the sources within the NSSS to those corresponding short-term curie releases that go with the 10 CFR 100 of 150 rem. One sees that the circulating activities of the iodines is very small. Similarly, the plateout. The expected activity after 40 years of operation is a factor of 10 lower, and even our design activity, which is more a 95 percent confidence number is two or three time lower than the 250 curies, which was the short-term release allowable.

So the purpose of this graph is to say that the main area of concentration, because of the fuel quality that we have that keeps the primary circuit clean, the main area of concentration is in the fuel particles and how to assure that during an event, those fuel particles remain intact.

DR. SIESS: Are the assumptions used in getting at those plateout numbers the same as those used at Fort St. Vrain?

MR. SILADY: They consider the experience we'cve had at Fort St. Vrain to a large degree. We've seen very good performance of the fuel there.

DR. SIESS: How much of a factor is that, compared with what was assumed in the FSAR for

Fort St. Vrain?

MR. SILADY: Actually, I believe in the FSAR for Fort St. Vrain, the plateout per pass was something like 40 times, and our experience is now that it is only 1 percent per pass. So these consider the new knowledge and new experience.

DR. SIESS: A factor of 40 then below Fort St. Vrain?

MR. SILADY: The actual curies around the circuit on a per megawatt basis are much less. The fuel quality we are specifying for this plant is approximately a factor of 5, better than what we have seen at Fort St. Vrain, which is another probably factor of 10. I've got the backup Vugraph here, better than the specification from Fort St. Vrain.

So we are requiring much tighter fuel quality than at Fort St. Vrain, and we are requiring fuel quality that is even better than what we've seen at Fort St. Vrain.

DR. SIESS: That gets back to my question. If I look at the column dated design activity plateout --

MR. SILADY: The 90 curies.

DR. SIESS: If I calculated that on the same basis as the FSAR for Fort St. Vrain, what would I get?

MR. SILADY: You would not get the same number of you did it in terms of per megawatt, okay, because we are specifying fuel quality.

DR. SIESS: I don't know whether you don't understand the question or don't want to answer it.

MR. SILADY: The former.

DR. SIESS: There were certain assumptions made at Fort St. Vrain about fuel quality, not specified, but assumptions, if you wish, or whatever, and plateout, et cetera, that led to some values for plateout. If you made those same assumptions, correcting for megawatts and all the other things, what would you get here instead of 80?

MR. SILADY: If you assumed the same assumptions and the same fuel quality and corrected for megawatts, you'd get 80.

DR. SIESS: You'd still get 80.

MR. SILADY: If you factor in that we're specifying better fuel, you get less than 80.

DR. SIESS: Okay. So 80 would be comparable to Fort St. Vrain, in terms of assumed fuel quality.

MR. SILADY: Excuse me. Maybe I misunderstood. One would not get 80. One would get a number higher than 80, if one used the same Fort St. Vrain assumptions, the same fuel quality as Fort St. Vrain and corrected for megawatts.

DR. CARBON: What would you get? About 400?

MR. SILADY: I don't know the number. I know it's at least an order of magnitude. Sorry.

DR. SIESS: So this is taking advantage of state of the art experience.

MR. SILADY: That's the point.

DR. SIESS: To get down to the 80. Now in the expected activity, do you have a comparable reduction, or is that a realistic comparison with Fort St. Vrain? There have been plateout measurements on Fort St. Vrain; right?

MR. SILADY: Both the plateout and the circulating activity has factored in the Fort St. Vrain experience.

I think since we've spent so much time on this one point, and since I didn't understand the question at the beginning, let me show you very briefly the comparison of the fuel quality of Fort St. Vrain versus the fuel quality that we're specifying here.

(Slide.)

This chart show the combined silicon carbide defect and contamination fraction versus essentially time, different cases, but there's an evolution here, and this is the Fort St. Vrain specification. Based on the experience that we saw at Fort St. Vrain for the 2240 megawatt plant, the larger plant, we specified the quality factor of 3 lower.

THTR in Germany has this criterion, 50 percent value. AVR has this quality of fuel with the 95 percent

upper bound value.

The two HRB designs, 500 megawatt HTR and their 100 megawatt, are at this point. Interatom is here. The fuel that Hobeg has made, has made here, here or here. Their criteria in Germany are exactly the same as we are not proposing on a 4 x 350, 6 x 10 to the minus 5 at the 50 percent confidence limit.

DR. SIESS: That's based on what you've actually been able to produce.

MR. SILADY: It's based on experience and what we've been able to produce.

I'd like to now then return to the next step in the process --

(Slide.)

-- which is to examine the design basis events. I am only going to go through one event.

DR. CARBON: Could I stop you just a minute. Your last comment was, your fuel specs are based on experience and what you've been able to produce. A lot of that is what you've been able to produce. How do you know you've been able to produce it? If I understand what you said, your experience comes from reactor operations, but you've gone well beyond that.

MR. SILADY: Yes. There are capsule radiation tests that are run on the particles and the silicon carbide

defects and contamination are measured.

DR. CARBON: Okay. Just a small number. Do you have a statistically significant amount?

MR. SILADY: The base is growing, but there has been considerable interchange between us and the Germans on a joint program exchanging capsules.

DR. CARBON: Even so, do you have a statistically significant amount yet? I don't know how to define that.

MR. NEYLAN: The Germans have produced statistical quantities of this quality of fuel, and as you saw, have produced test quality fuel, which is considerably better than the specifications. The U.S. has not produced this reference fuel in statistical quantities. It has, of course, produced Fort St. Vrain, which is the basis of our experience. The Germans have also used production in AVR. That is their commercial offering at the same specification as we are offering.

MR. SILADY: Thank you.

The one event I do want to go through is defined in this fashion.

It begins with a moderate, that is, a very small primary coolant leak. Moderate is a relative term, but this is less than a square inch. The reactor is tripped, based on high radioactivity in the reactor building or low pressure in the primary circuit. The specifics of this

event is that force core cooling is lost. You recall from Neylan's presentation that we can remove the decay heat, not only with the heat transport system but with the shutdown cooling system, but in this particular event, both of those are lost.

The decay heat that is removed via conduction and radiation to the reactor cavity cooling, and it's only with conduction radiation and not convection in this particular event, because we are depressurizing, because of the leak, circulating activity is then released, and we get an incremental release from the fuel, as the core temperatures exceed normal operating levels.

Let me show you those temperatures.

DR. SIESS: Is this the worst case for Fort St. Vrain, no pressure, no circulation?

MR. SILADY: Yes; that's known as Design Basis Accident No. 1 Fort St. Vrain.

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(Slide.)

This diagram shows the traditional characteristics of HTGR's slow time response. The chart goes to 1000 hours and extends from the normal operating range of the fuel, which on average is roughly 700 C but the peak could be at 1100 C, and one sees that temperatures do go up in order for the heat to be conducted and radiated out through the side of the vessel. But at about 80 hours the temperatures turn over and come down.

The average temperature is considerably lower than the maximum. This is shown in the next graph --

(Slide.)

-- via an isotherm map on a cutaway and RZ geometry.

L me orient you.

This is the axial direction. This is the radial direction.

The active core, that which has the fuel, then is within this annular region, which shows up here as a rectangle.

The peak temperature occurs just on the inside boundary near the graphite center column.

Note that much of the core is considerably below 1600 C and since volumetrically this out here contributes much more to the average temperature, the average

temperature is even lower than one would first see by looking at this.

The side reflector continues to be lower in temperature, and the normal gradient with the heat being transferred through the vessel and out to the reactor cavity cooling system.

(Slide.)

Repeating the viewgraph that Tony Neylan showed you, the fission gas release from particles is seen to be negligible up to very high temperatures, when the silicon carbide thermally begins to degrade. This is one component of a release then from the core during this event. No incremental failure essentially.

Other components are the release from the contamination or initially failed particles before the event occurs. All of those contributions are taken into account.

(Slide.)

And the result over time is the following graph, where now the time scale is still 1000 hours and these are cumulative Iodine-131 in curies. Iodine-131 contributes about 80 percent of all the iodines and therefore 80 percent of the thyroid dose. One sees a very slow release then.

And the top graph, the solid one is from the fuel. Its peak value, which occurs hundreds of hours out into the event, is roughly 400 curies compared again to our

long-term allowable or corresponding curies to the 150 rem. One sees some margin there.

That, basically, is the approach then to removing core heat and retaining the fission products --

(Slide.)

-- within the fuel.

DR. CARBON: Is this for the case where you lose all forced convection and you shut down?

MR. SILADY: Yes.

DR. CARBON: There are no active components?

MR. SILADY: None.

There are many other accidents -- and we could have a day's session on looking at the cases -- when it is pressurized or different variations. But I went to one that shows you the functions and is going to show you the design selections that were intentionally chosen in order to meet our top level regulatory criteria.

If one looked at other events, one would find that there are two other functions that are needed in order to keep the fission products within the fuel. That is control reactivity or heat generation and control chemical attack.

And so by looking at a spectrum of events, examining those events, one finds the functions, and one can drive this down further. In term of perhaps for controlling

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reactivity, one needs to maintain core geometry, for instance.

That is the kind of thing we are doing, and again this is the basis for developing the principal design criteria.

I would like to now move into the third and final section, which is equipment classification. I would like to start again with some definitions.

(Slide.)

Safety-related structures, systems, and components are those performing these functions that we rely upon in order to meet 10 CFR 100 for the design basis events.

So you see again this is tied to the other two parts.

The safety-related SSEs are described and their limiting design conditions; that is, the conditions under which the safety-related equipment alone can keep you within 10 CFR 100, are evaluated in the PSID.

There are three steps in this third and final part of the methodology.

(Slide.)

The first is one I will go into a little bit more than the other two, to start with each design basis event and classify as safety related those systems, structures or

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components that are needed for compliance with the dose criteria of 10 CFR 100.

(Slide.)

Returning to our risk plot and to the point for which I showed you the transients, this particular event here, CC-sub-P-dash-9. If it were not for the radionuclide retention function of controlling the release, the point would be out here and we would violate 10 CFR 100. So we saw that we had to remove core heat and keep the fission products within the fuel particles.

So our design selections in this case, then, are some form of alternate core cooling and the fuel quality itself.

Let me go into another layer of depth, then.

(Slide.)

You have already seen this in terms of the first two, the specific numbers in that backup viewgraph of the design selection that we are specifying as safety related.

There is also a coating failure fraction; that is, one can back out -- that would be allowable during normal operation and transients similarly -- to keep within those curies that corresponds to 10 CFR 100.

The fuel quality, though, was not the only thing that had to be done.

(Slide.)

One also had to be able to remove the core heat. These are the design selections, then, that were made that would become safety related in order to do that.

The core power density is quite low, 5.9 watts per cc. The annular geometry at this power level, 1.65 meters in inner diameter and 3.5 outer. Graphite, with its heat capacity and its conductivity properties -- this should read low alloy steel -- that is a vessel that the heat can conduct out through. And a heat sink, which in this case we have chosen to be the reactor cavity cooling.

I would like now to talk about why we chose this decay heat removal system to be the one that is safety-related.

(Slide.)

You can imagine that for each of the functions there are corresponding processes and examination of events. I am taking just one slice through it as an example.

But for the function to remove core heat, one examines the spectrum of design basis events in which this function must operate in order to be within 10 CFR 100, and one looks at all the different systems that are available to perform that function.

So these are the system structures and components that are available -- in this case for heat removal, mostly

systems to remove decay heat in this particular first event.

But I didn't go into the design basis event one. All of these were available. Each of them alone could have removed the decay heat.

So the "yes" is, yes, it is available to perform the decay heat removal function. Not all of them are needed. Any one of them could have done it.

In design basis event five, it started off with the loss of the main loop. So it wasn't available.

In design basis event eight there was only one of these systems that could perform this function to the level required, and it was the reactor cavity cooling in the passive mode.

What one does then is look across and make a choice of which system or systems one makes safety related in order to satisfy this function, in order to satisfy the top level regulatory criteria of 10 CFR 100 for design basis events.

This example is a kind of simplified. It just works out that there is only one choice that covered the whole spectrum of events. So the reactor cavity cooling was chosen as safety related.

DR. SIESS: Do you only have one system, then?

MR. SILADY: The entire plant, which does all the

functions and satisfies all the requirements of the user and the regulatory body includes all these systems.

DR. SIESS: If I look at DBE-8, you then say one system to remove decay heat is enough.

MR. SILADY: In each of these events, if this is the only system you have, it is sufficient to keep the doses within 10 CFR 100.

DR. SIESS: I am sorry. I am looking at DBE-8, where there is only one "yes" in the column. Now, I have one system.

Where is the defense in depth?

MR. SILADY: The defense in depth is in the fact that one has the fuel particles and the graphite core and the other characteristics in order to provide backups to the reactor cavity cooling.

DR. SIESS: If that system fails, what does that do to your point on your curve, on your plot?

MR. SILADY: This is the system that we need.

DR. SIESS: And if it fails?

MR. SILADY: I believe the specifics of that event are that you start off pressurized and you have a loss of the main loop and you have a loss of the shutdown cooling system.

One then needs to have the reactor cavity cooling removing the energy in that case because the system is

pressurized.

If one does not have that and if one does not somehow repair whatever went wrong with the passive system, the operator can intentionally depressurize the system, in which case it is very similar to these other design basis events, where even dissipation of heat to the ground is sufficient.

DR. SIESS: So removing that capability moves you down?

MR. SILADY: Off the range.

DR. SIESS: Of the probability curve and potentially could move you over on the dose curve?

MR. SILADY: That event doesn't show up in the design basis region.

DR. SIESS: But it would push you into that other region in terms of probability?

MR. SILADY: That is correct.

DR. SIESS: And of course, I assume that system itself has redundant components?

MR. SILADY: Yes. You will see this afternoon the level of redundancy. Yes.

DR. SIESS: Okay.

(Slide.)

MR. SILADY: The final two steps, just briefly.

One has to make sure that events are kept within

10 CFR 100 of the design basis events, and one specifies the function and the design selection then as safety related.

Similarly, you wouldn't want to have an event lower in frequency with doses greater than 10 CFR 100 that could, if it were not made safety related, cause you to exceed the 10 CFR 100. So one wants to keep that event out of the design basis region.

So this step reads that for each emergency planning basis event with consequences greater than 10 CFR 100 classify as safety related those design selections that are needed to assure that it is low in frequency, that it is below the design basis region.

The final step is to classify as safety related for each design selection classified as safety related to determine the limiting design conditions for its operation by examining all its associated DBEs.

That gets back to your question. We would look at all those design basis events, the one that was pressurized and the others that were depressurized, and calculate the way in which the reactor cavity cooling system has to respond. That becomes its limiting design condition.

I would like to summarize then. I don't know if I have made any time, as I tried to do. But here is our summary.

(Slide.)

The method uses the functional analysis of the integrated approach, PRA, the risk assessment, and reliability tools and the resulting design selections to show compliance with the top level regulatory criteria. The key is it always is linked back into that top level regulatory criteria.

The method provides a systematic, traceable process to derive the licensing bases specific to the modular HTGR.

The application of the method demonstrates that the modular HTGR is emphasizing retention within the fuel with passive features.

And the approach is consistent with the draft advanced reactor policy.

Any additional questions?

DR. SIESS: I am still thinking in terms of your frequency consequence diagrams. I know there is always room for argument and uncertainty about reliability and of the probability curve, and I guess until we got into this light water reactor source term I thought there was less argument about what the consequences might be, and I still have some hope that for the light gas cooled reactors there might be less argument.

Can you conceive of a maximum release without

DAVbur

assigning a probability to it?

MR. SILADY: In essence, that is what we are doing here. We are trying to come up with a scheme that guarantees that the doses are below the protective action guidelines with a design that performs those functions for a wide spectrum of events, going all the way down to 5 times 10 to the minus 7th.

DR. SIESS: Right now it seems to me that if you scram the reactor and you can't dump the stuff into the earth without a circulating system, you can't get too much out of there in the way of fission products.

MR. SILADY: That is correct. The design is set up so that it can remove the heat passively without releasing from the fuel particles.

DR. SIESS: To get more than that out you would have to have a failure to scram?

MR. SILADY: I don't even know if that is that crucial.

We have taken a look at the negative temperature coefficient of the system and looked at the event I have just showed you plus others without trip, and the temperatures go up because it takes a few minutes in order for the negative temperature coefficient to turn the transient around. But they only go up by 50 degrees, or something.

So essentially our releases are the same.

I didn't get off into those events and get off into controlling heat generation today because I was really trying to get the feedback on the approach.

But in another session we could go into the high degree of passivity that is in the design for controlling heat generation.

DR. SIESS: Let me say that I think you have got -- the approach is quite transparent -- questions about it.

I understand it. I think it is a very helpful way of looking at it, and by being transparent that simply opens you to a lot more questions.

(Laughter.)

DR. SIESS: But I think that is an object of the game.

MR. SILADY: Thank you very much.

(Whereupon, at 12:55 p.m., the subcommittee was recessed, to go into closed session.)

CERTIFICATE OF OFFICIAL REPORTER

This is to certify that the attached proceedings before the UNITED STATES NUCLEAR REGULATORY COMMISSION in the matter of:

NAME OF PROCEEDING: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON ADVANCED REACTORS

DOCKET NO.:

PLACE: Washington, D. C.

DATE: Thursday, January 30, 1986

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

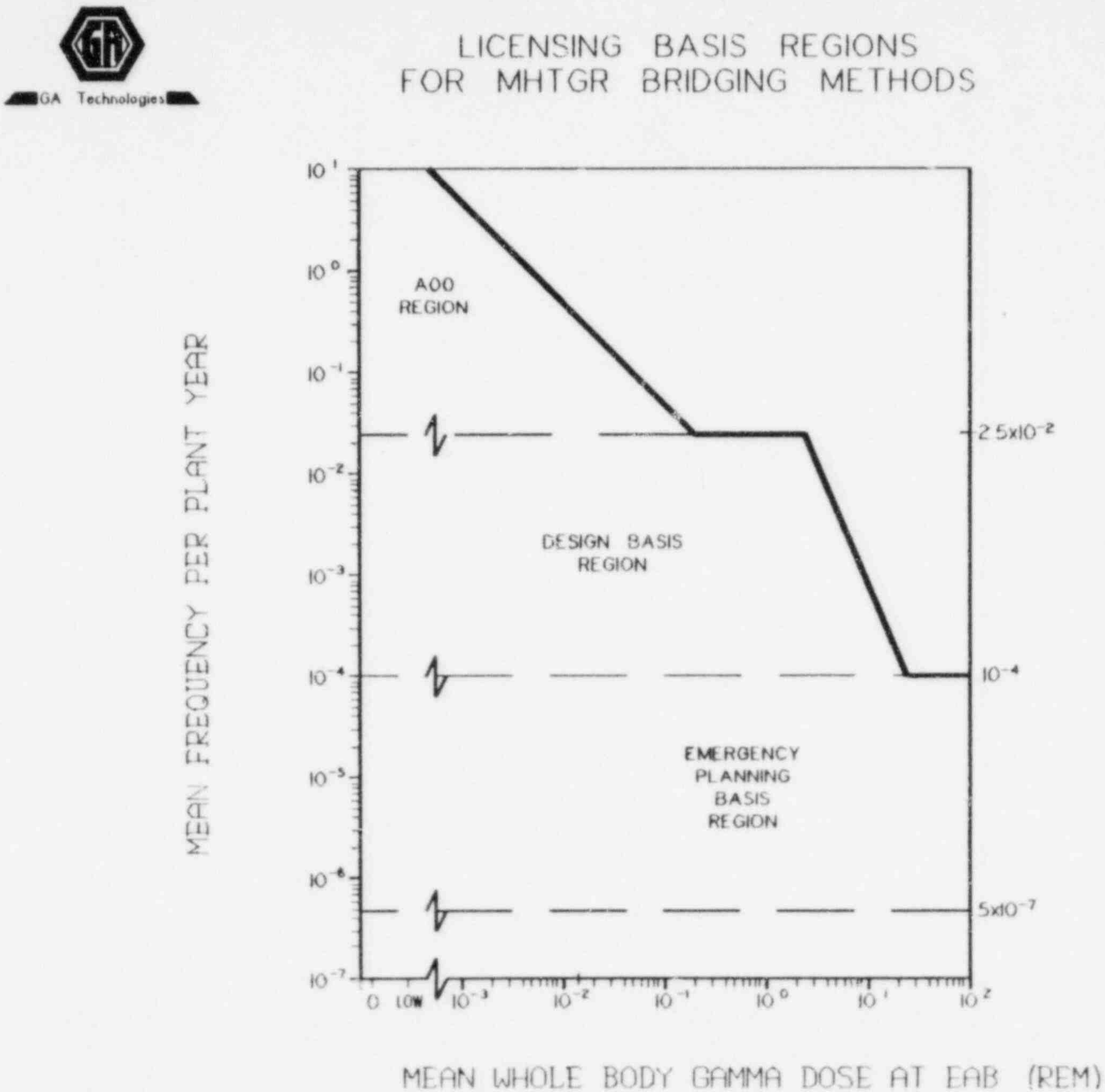
(sigt) David L. Hoffman

(TYPED)

DAVID L. HOFFMAN
Official Reporter

ACE-FEDERAL REPORTERS, INC.
Reporter's Affiliation

LICENSING BASIS REGIONS FOR MHTGR BRIDGING METHODS



MHTGR

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LICENSING BASIS EVENT SELECTION

STEP 1:

DEFINE THREE REGIONS ON A FREQUENCY–CONSEQUENCE RISK PLOT

STEP 2:
COMPARE RISK ASSESSMENT OF THE DESIGN TO THE FREQUENCY CONSEQUENCE RISK PLOT

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EVENT DEFINITIONS (CONTINUED)

EMERGENCY PLANNING BASIS EVENTS (EPBEs)

EVENTS OF LOWER FREQUENCY THAN DBEs, NOT EXPECTED TO OCCUR IN THE LIFETIME OF ALL STANDARD HTGRs

CONSEQUENCES REALISTICALLY ANALYZED IN PRA AND CHAPTER 13 OF SARs (PSID) FOR COMPLIANCE WITH PAG DOSE LIMITS

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EVENT DEFINITIONS (CONTINUED)

DESIGN BASIS EVENTS (DBEs)

EVENTS OF LOWER FREQUENCY THAN AOOs, NOT EXPECTED TO OCCUR IN THE LIFETIME OF THE PLANT

CONSEQUENCES CONSERVATIVELY ANALYZED IN CHAPTER 15 OF SARs (PSID) FOR COMPLIANCE WITH 10CFR100

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PRESENTATION OUTLINE

- LICENSING BASIS EVENTS
  - –DEFINITIONS
  - –METHOD / APPLICATION
- REQUIRED FUNCTIONS
  - –DEFINITIONS
  - –METHOD
  - –APPLICATION
- EQUIPMENT CLASSIFICATION
  - –DEFINITIONS
  - –METHOD / APPLICATION
- SUMMARY

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DEVELOPMENT OF LICENSING BASES

PRESENTED TO THE ACRS

JANUARY 30, 1986

FRED A. SILADY

GA TECHNOLOGIES, INC.

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EVENT DEFINITIONS (CONTINUED)

ANTICIPATED OPERATIONAL OCCURRENCES (AOOs)

EVENTS EXPECTED ONCE OR MORE IN THE PLANT LIFETIME

CONSEQUENCES REALISTICALLY ANALYZED IN CHAPTER 11 OF SARs (PSID) FOR COMPLIANCE WITH 10CFR50 APPENDIX I

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LICENSING BASIS EVENT DEFINITIONS

<u>LICENSING</u> <u>BASIS</u> <u>EVENTS</u> (LBEs)

THE OFF–NORMAL OR ACCIDENT EVENTS USED FOR DEMONSTRATING DESIGN COMPLIANCE WITH THE TOP–LEVEL REGULATORY CRITERIA

COLLECTIVELY LBEs ARE ANALYZED IN PRAs FOR DEMONSTRATING COMPLIANCE WITH INTERIM SAFETY RISK GOALS

LBEs ENCOMPASS THE FOLLOWING THREE EVENT CATAGORIES

LICENSING APPROACH SUMMARY

THREE STEPS:

1) IDENTIFY TOP-LEVEL CRITERIA GENERIC TO ALL REACTOR TYPES AS STARTING POINT.

   - DONE.

2) DEVELOP PROCESS TO DERIVE LICENSING BASES SPECIFIC TO THE MHTGR WHICH ENSURE THAT TOP-LEVEL CRITERIA ARE MET.

   - DONE.

3) APPLY PROCESS TO IDENTIFY MHTGR LICENSING BASES.

   - IN PROGRESS.

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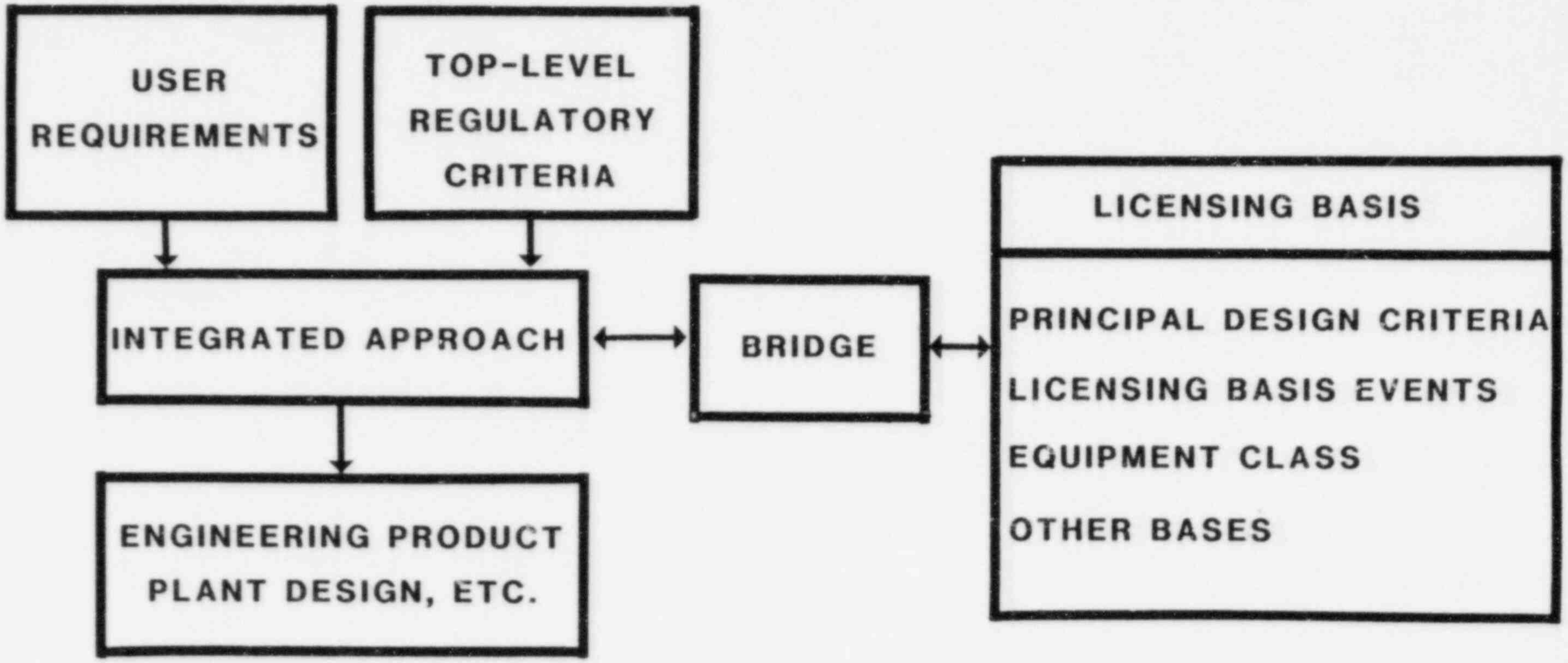
LICENSING METHODOLOGY SUMMARY

- USE PRA TO IDENTIFY LIKELIHOOD OF EVENTS AND CLASSIFY EVENTS INTO THREE REGIONS FOR COMPARISON AGAINST TOP LEVEL REGULATORY CRITERIA
- EXAMINE EVENTS TO IDENTIFY REQUIRED FUNCTIONS TO MEET THE TOP LEVEL REGULATORY CRITERIA
- CHOOSE DESIGN SELECTIONS TO ACCOMPLISH REQUIRED FUNCTIONS TO MEET THE TOP LEVEL REGULATORY CRITERIA

HTGR DESIGN & LICENSING APPROACH



OVERVIEW OF THE LICENSING BASES DEVELOPMENT

PRESENTED TO THE ACRS

JANUARY 30, 1986

FRED A. SILADY

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MHTGR PRELIMINARY SAFETY RISK ASSESSMENT COMPARED TO APPENDIX I AND 10CFR100

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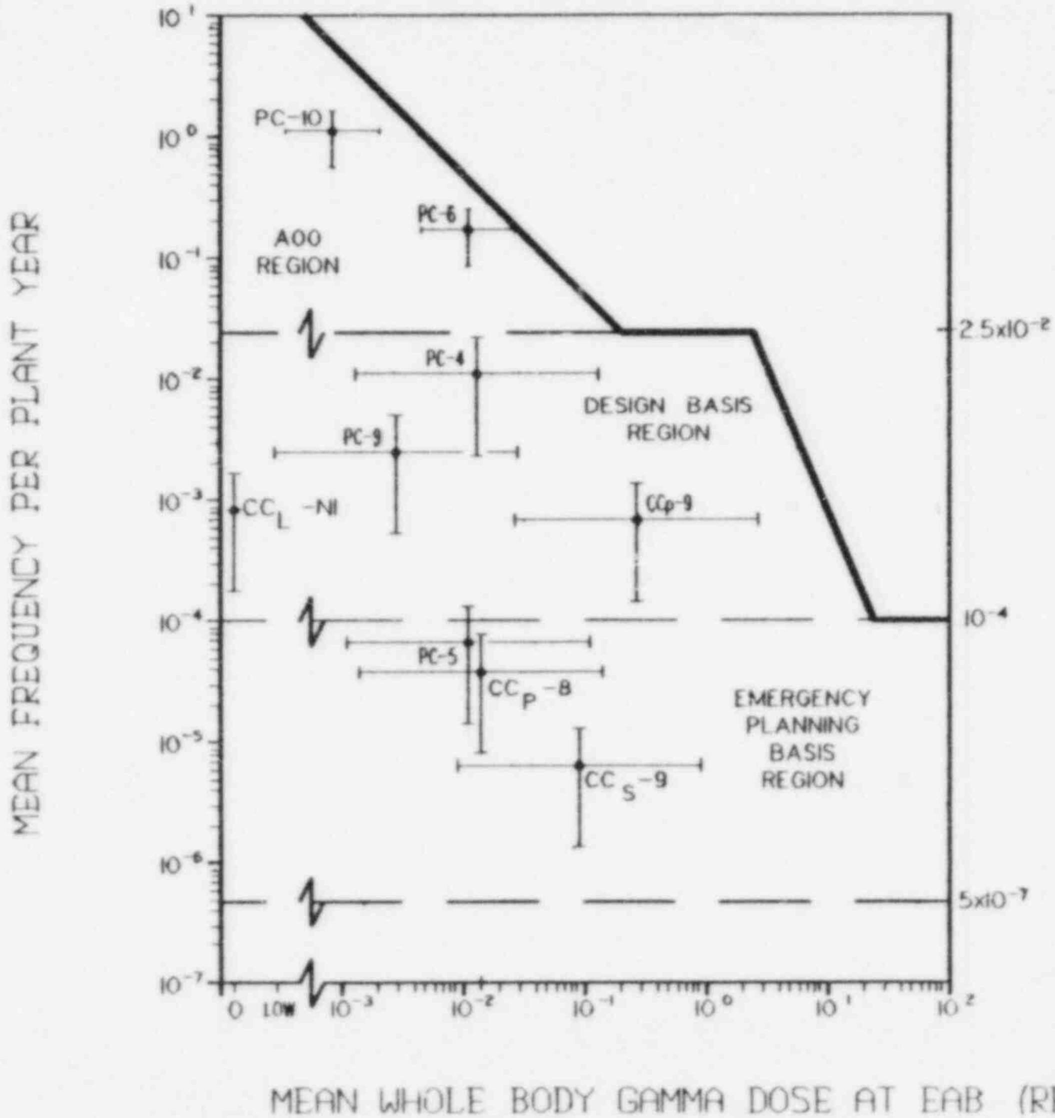

MHTGR5

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AAO SELECTION / EVALUATION

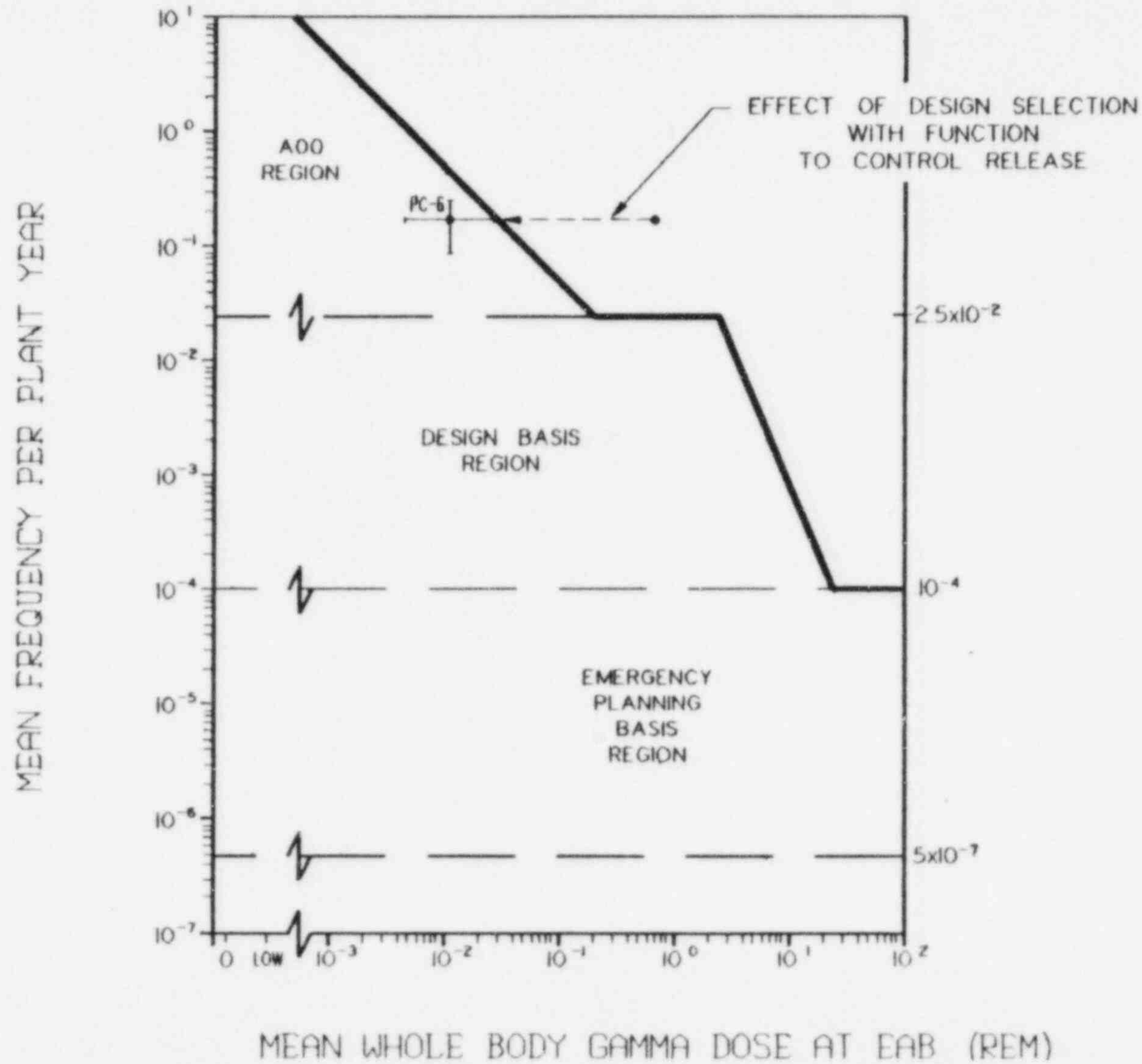
STEP 3:

IDENTIFY AS AAOs THOSE FAMILIES OF EVENTS WITHIN THE AAO REGION

STEP 4:

EVALUATE THE CONSEQUENCES OF THE SELECTED AOOs REALISTICALLY

IDENTIFY PC-6 AS ANTICIPATED OPERATIONAL OCCURRENCE



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DESIGN BASIS EVENT SELECTION / EVALUATION

STEP 5:

IDENTIFY AS DBEs THOSE FAMILIES OF EVENTS WITHIN THE DESIGN BASIS REGION

STEP 6:

IDENTIFY AS DBEs THOSE EVENTS WITH AGREED UPPER OR LOWER BOUND FREQUENCIES THAT LIE WITHIN THE DESIGN BASIS REGION AND SATISFY STEP 5

STEP 7:

EVALUATE THE CONSEQUENCES OF THE SELECTED DBEs CONSERVATIVELY

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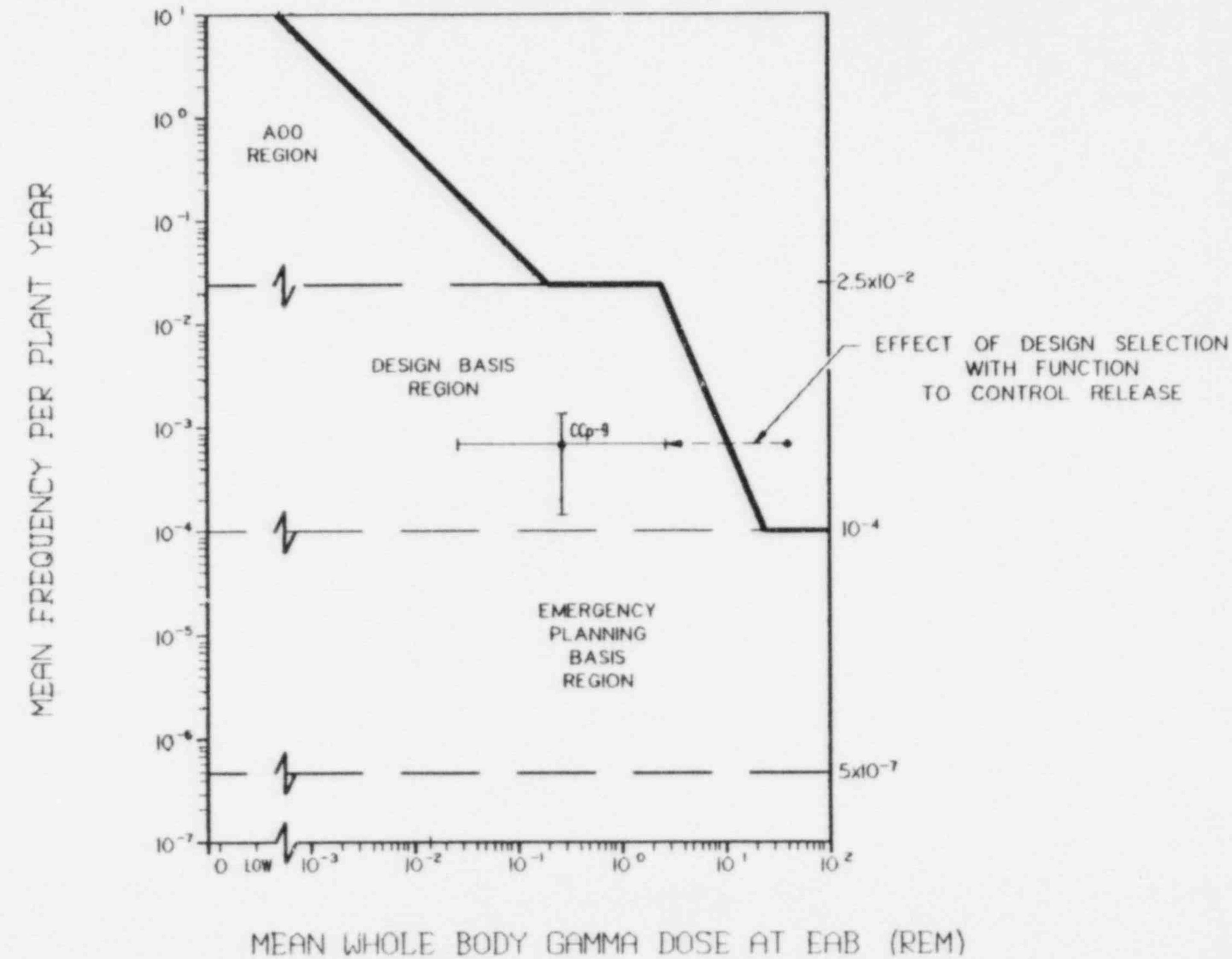
EXAMPLE SELECTION OF DESIGN BASIS EVENT

DEPRESSURIZED CONDUCTION COOLDOWN WITH RCCS
CC_P–9

EVENT SEQUENCE:

1. MODERATE PRIMARY COOLANT LEAK OCCURS
2. REACTOR IS TRIPPED
3. FORCED CORE COOLING IS LOST (BOTH HTS AND SCS)
4. DECAY HEAT REMOVED VIA CONDUCTION AND RADIATION TO REACTOR CAVITY COOLING
5. CIRCULATING ACTIVITY IS RELEASED
6. INCREMENTAL RELEASE FROM FUEL AS CORE TEMPERATURES EXCEED NORMAL OPERATING LEVELS

IDENTIFY CC_p−9 AS DESIGN BASIS EVENT



CCP9

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EPBE SELECTION / EVALUATION

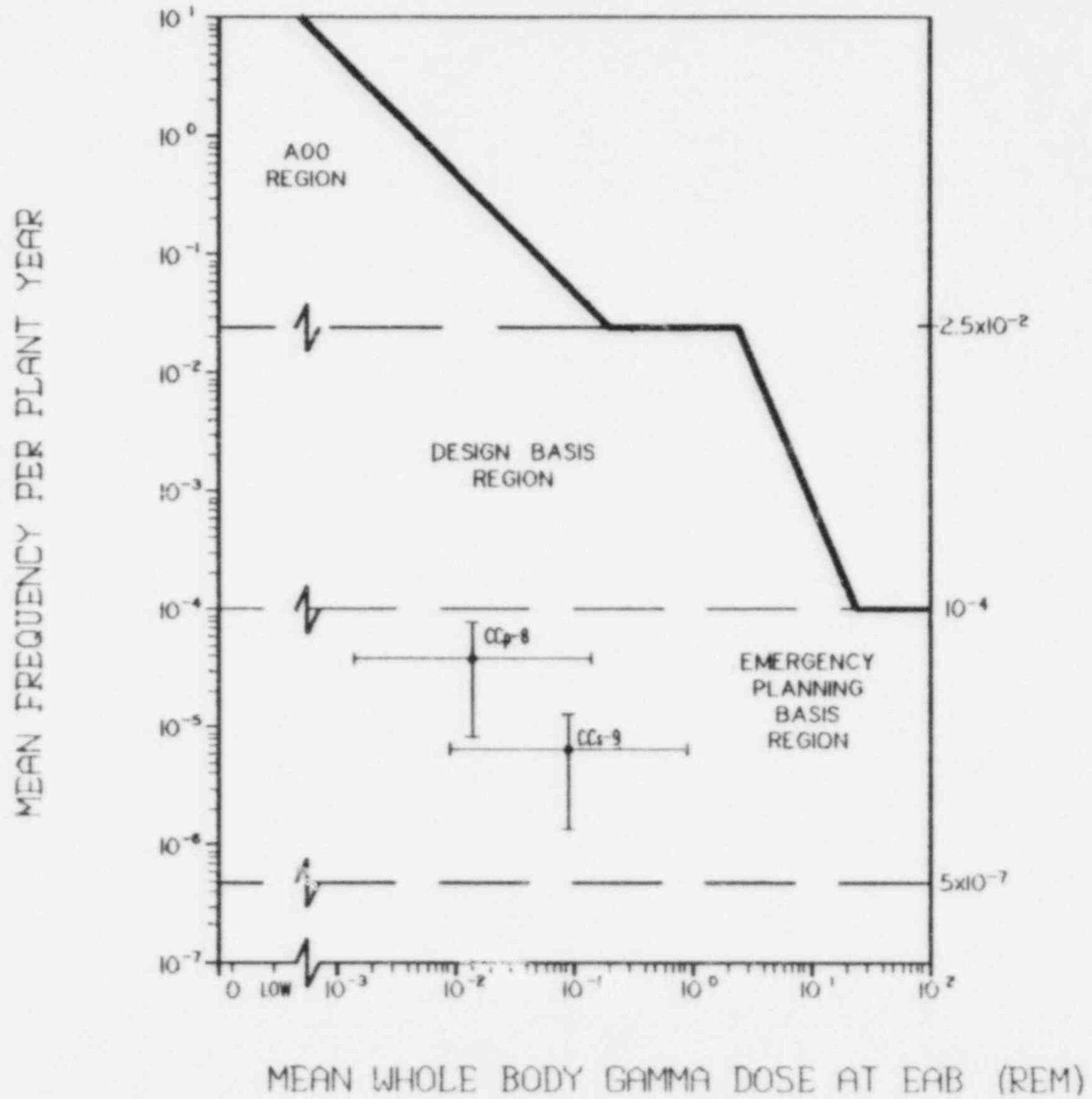
STEP 8:

IDENTIFY AS EMERGENCY PLANNING BASIS EVENTS (EPBEs) THE CONSEQUENCE–DOMINANT EVENTS WHOSE UPPER BOUND FREQUENCY IS WITHIN THE EMERGENCY PLANNING BASIS REGION

STEP 9:

EVALUATE THE CONSEQUENCES OF THE SELECTED EPBEs REALISTICALLY

IDENTIFY CC_s-9 AS EMERGENCY PLANNING BASIS EVENT



CCS9

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OVERALL REGULATORY COMPLIANCE

STEP 10:

COMPARE THE RISK ASSESSMENT OF THE INTEGRATED APPROACH DESIGN TO THE INTERIM RISK GOALS

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COMPARISON OF PRELIMINARY ASSESSMENT TO LATENT CANCER RISK GOAL

REGION	EVENT	EVENTS PER PLANT YEAR	CANCERS PER EVENT	RISK
AOO	PC-10	1.1	5×10^{-9}	6×10^{-9}
AOO	PC-6	0.17	7×10^{-8}	1.2×10^{-8}
10CFR100	PC-4	1×10^{-2}	1×10^{-7}	1×10^{-9}
10CFR100	PC-9	2×10^{-3}	2×10^{-8}	4×10^{-11}
10CFR100	CC_p-9	7×10^{-4}	2×10^{-6}	1×10^{-9}
EPB	PC-5	7×10^{-5}	1×10^{-7}	7×10^{-12}
EPB	CC_p-8	4×10^{-5}	1×10^{-7}	4×10^{-12}
EPB	CC_s-9	6×10^{-6}	7×10^{-7}	4×10^{-12}
			TOTAL RISK	2×10^{-8}
			RISK GOAL	2×10^{-6}
			MARGIN	100X

REQUIRED FUNCTIONS

- DEFINITIONS
- METHOD
- APPLICATION

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REQUIRED FUNCTIONS DEFINITION

THOSE FUNCTIONS NEEDED TO LIMIT RADIONUCLIDE RETENTION TO MEET 10CFR100 DOSES FOR DBEs

BASIS FOR PRINCIPAL DESIGN CRITERIA EVALUATED IN CHAPTER 3 OF SARs (PSID) FOR COMPLIANCE WITH TOP LEVEL REGULATORY CRITERIA

SELECTION OF REQUIRED FUNCTIONS

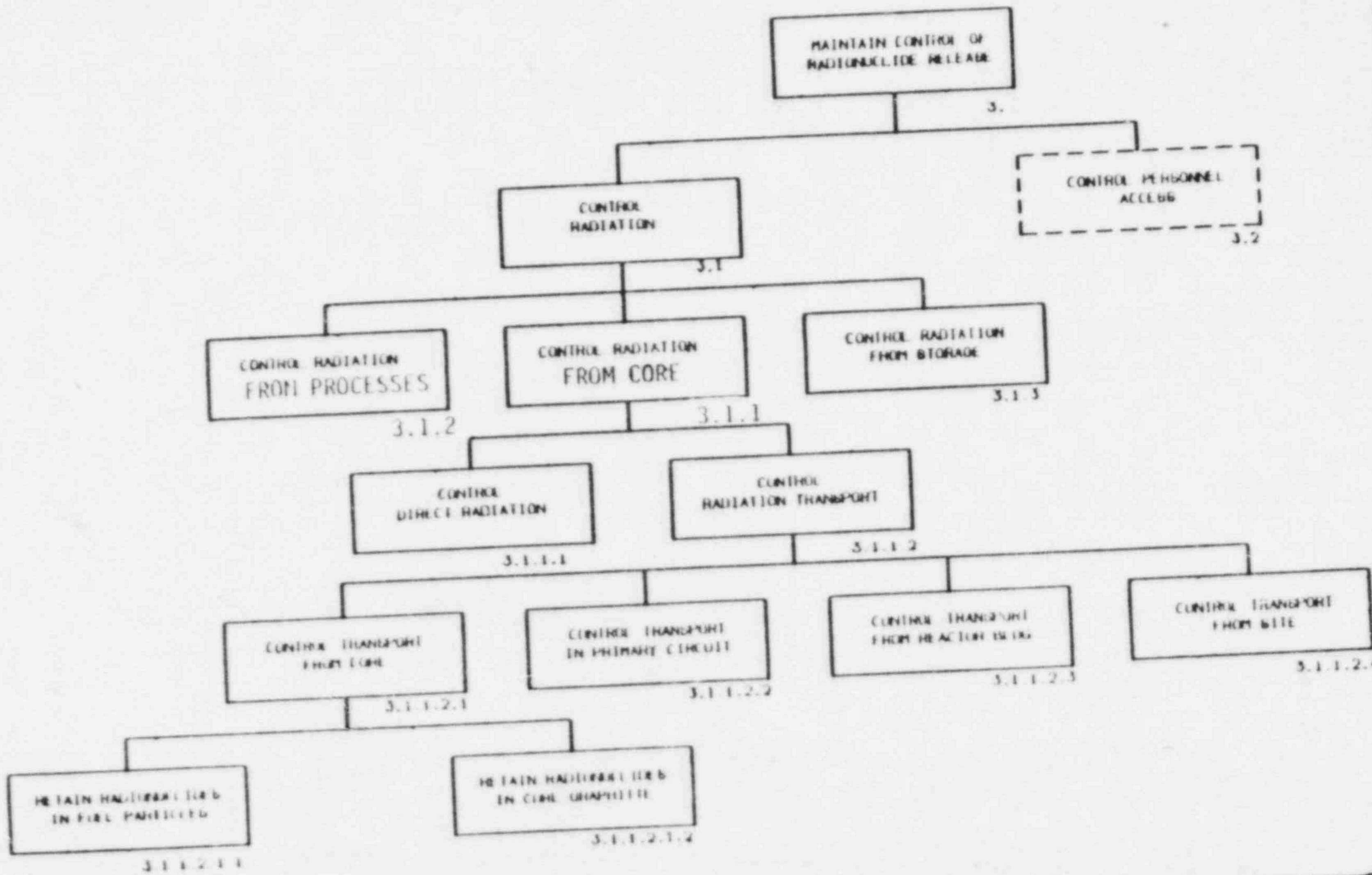
STEP 1:

IDENTIFY THOSE RADIONUCLIDE RETENTION FUNCTIONS REQUIRED TO MEET TOP LEVEL REGULATORY CRITERIA AND USER/UTILITY SAFETY REQUIREMENTS

STEP 2:

SELECT THOSE RADIONUCLIDE RETENTION FUNCTIONS REQUIRED TO MEET 10CFR100 DOSES FOR DBEs

FUNCTIONS REQUIRED TO MAINTAIN CONTROL OF RADIONUCLIDE RELEASE



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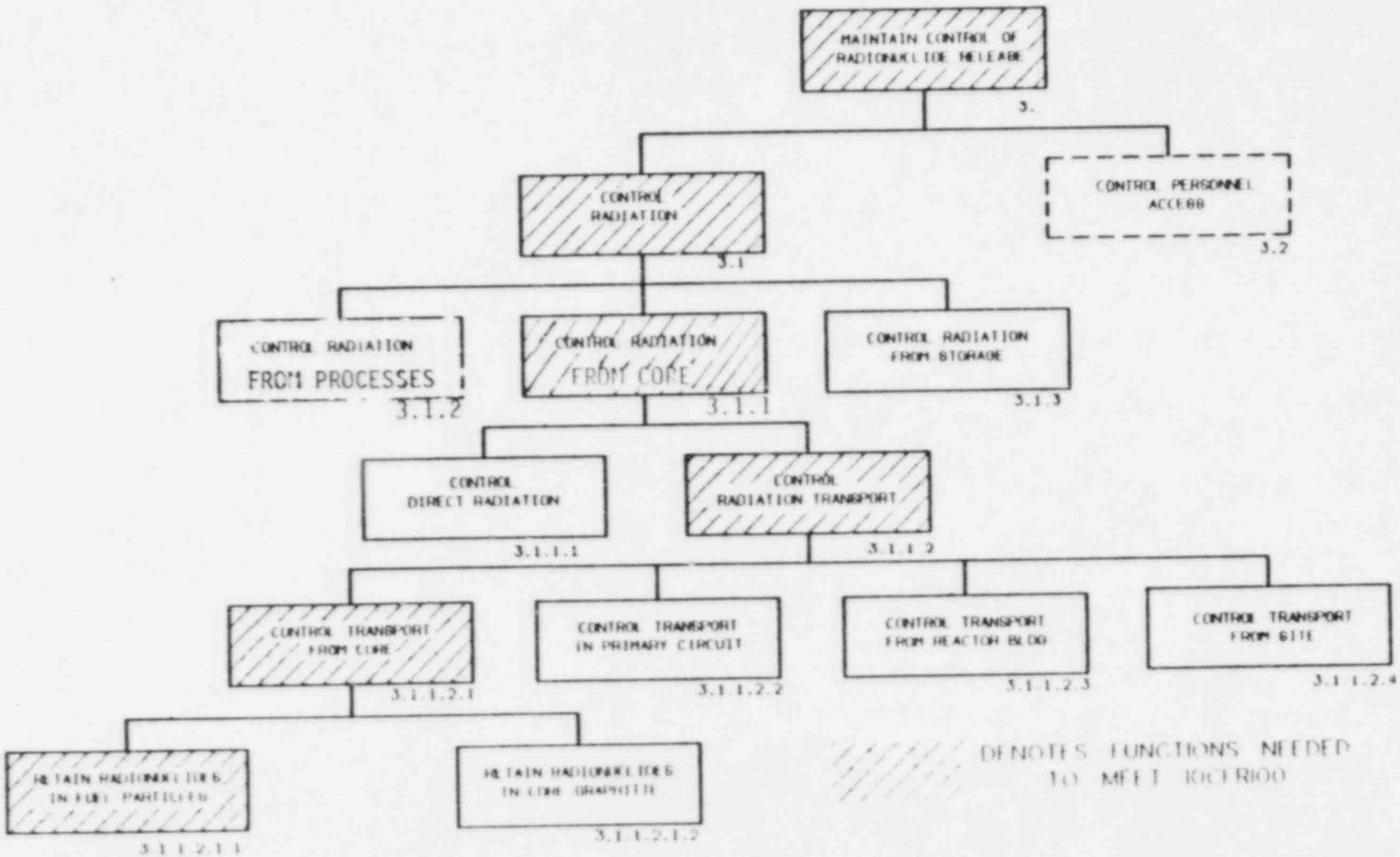
FUNCTIONS REQUIRED TO MAINTAIN CONTROL OF RADIONUCLIDE RELEASE TO WITHIN 10CFR100 DOSES



FT3A

IODINE RELEASE CORRESPONDING TO THYROID 10CFR100 DOSE

	SHORT TERM RELEASE	LONG TERM RELEASE
TIME FRAME	0 - 8 HOURS	0 - 30 DAYS
ATMOSPHERIC DISPERSION	1.22×10^{-3} Sec/m^3	5.11×10^{-4} Sec/m^3
BREATHING RATE	3.47×10^{-4} m^3/Sec	2.32×10^{-4} m^3/Sec
DOSE CONVERSION FACTOR	1.49×10^{6} Rem/Ci	1.49×10^{6} Rem/Ci
IODINE CURIES CORRESPONDING TO 10CFR100 150 Rem	250	900

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COMPARISON OF I131 SOURCES TO ALLOWABLE RELEASES

SOURCE	EXPECTED ACTIVITY Ci	DESIGN ACTIVITY Ci	10CFR100 Ci
CIRCULATING	.02	.08	250
PLATEOUT	20	80	250

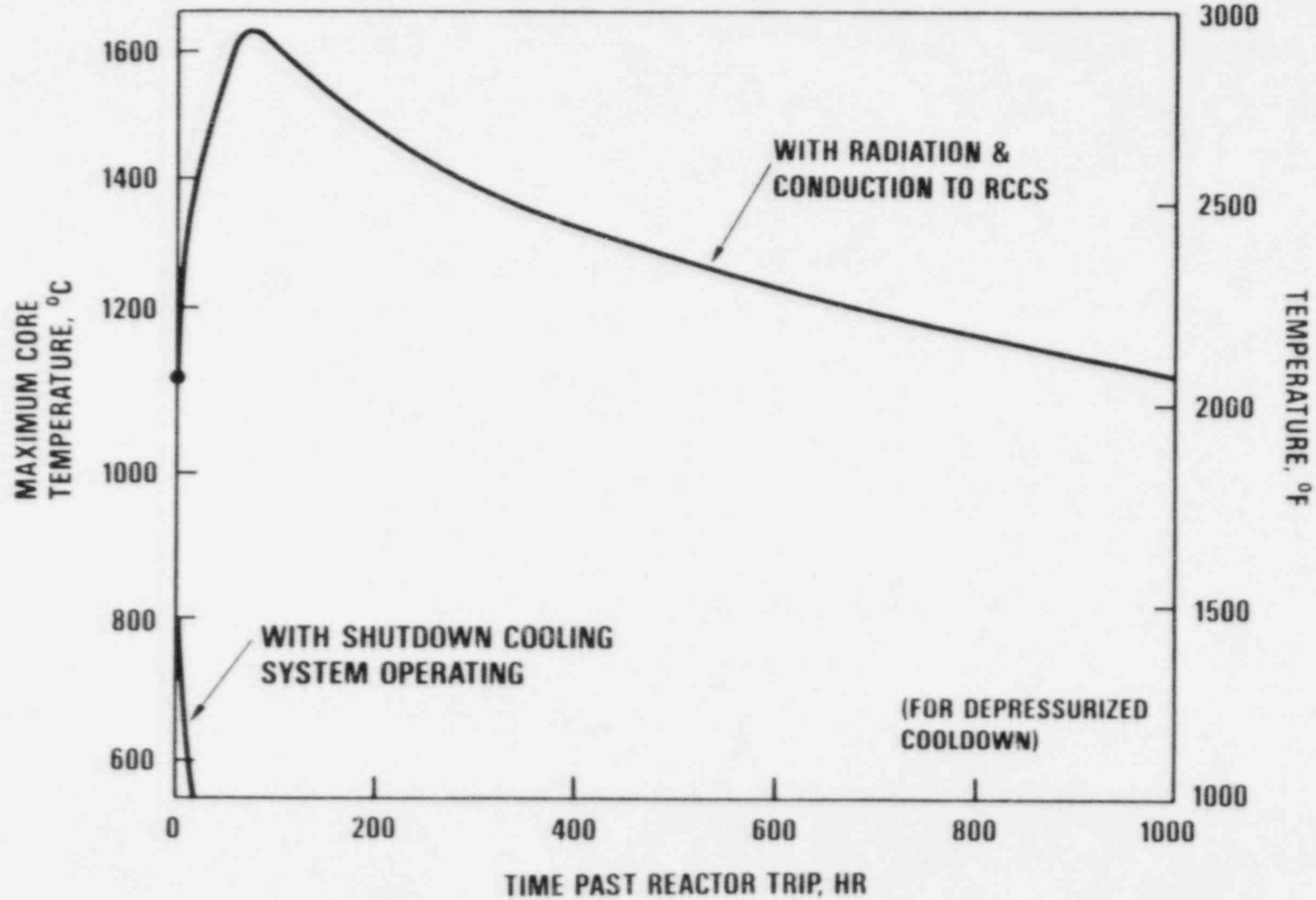
EXAMPLE DBE EXAMINATION FOR REQUIRED FUNCTIONS

DEPRESSURIZED CONDUCTION COOLDOWN WITH RCCS
CC_P–9

EVENT SEQUENCE:

1. MODERATE PRIMARY COOLANT LEAK OCCURS
2. REACTOR IS TRIPPED
3. FORCED CORE COOLING IS LOST (BOTH HTS AND SCS)
4. DECAY HEAT REMOVED VIA CONDUCTION AND RADIATION TO REACTOR CAVITY COOLING
5. CIRCULATING ACTIVITY IS RELEASED
6. INCREMENTAL RELEASE FROM FUEL AS CORE TEMPERATURES EXCEED NORMAL OPERATING LEVELS

CORE TEMPERATURES REMAIN BELOW DESIGN LIMITS DURING ACCIDENTS



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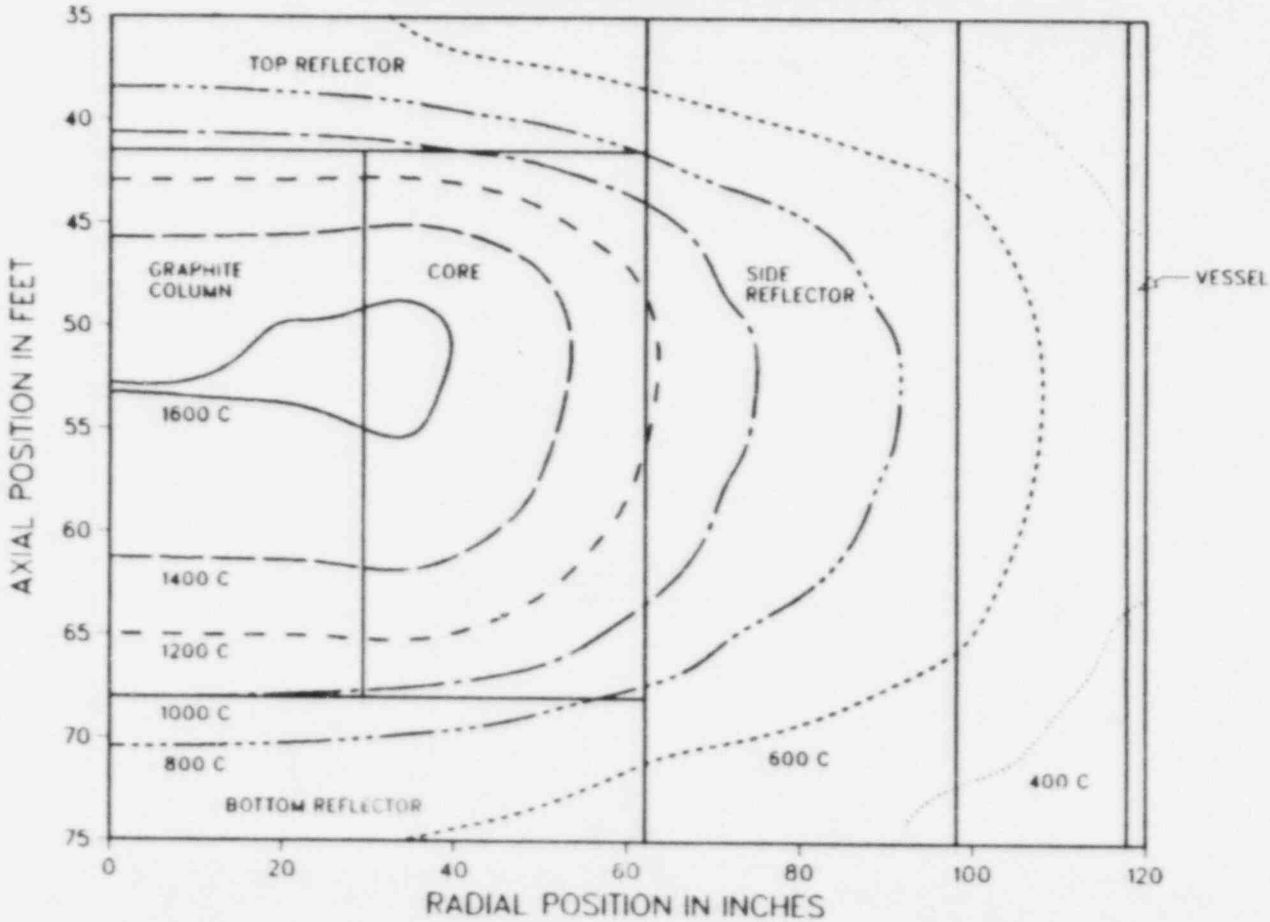
ISOTHERM MAP AT TIME OF PEAK (79 HRS) CORE TEMPERATURE FOR 350 MW(t) DEPRESSURIZED CONDUCTION COOLDOWN



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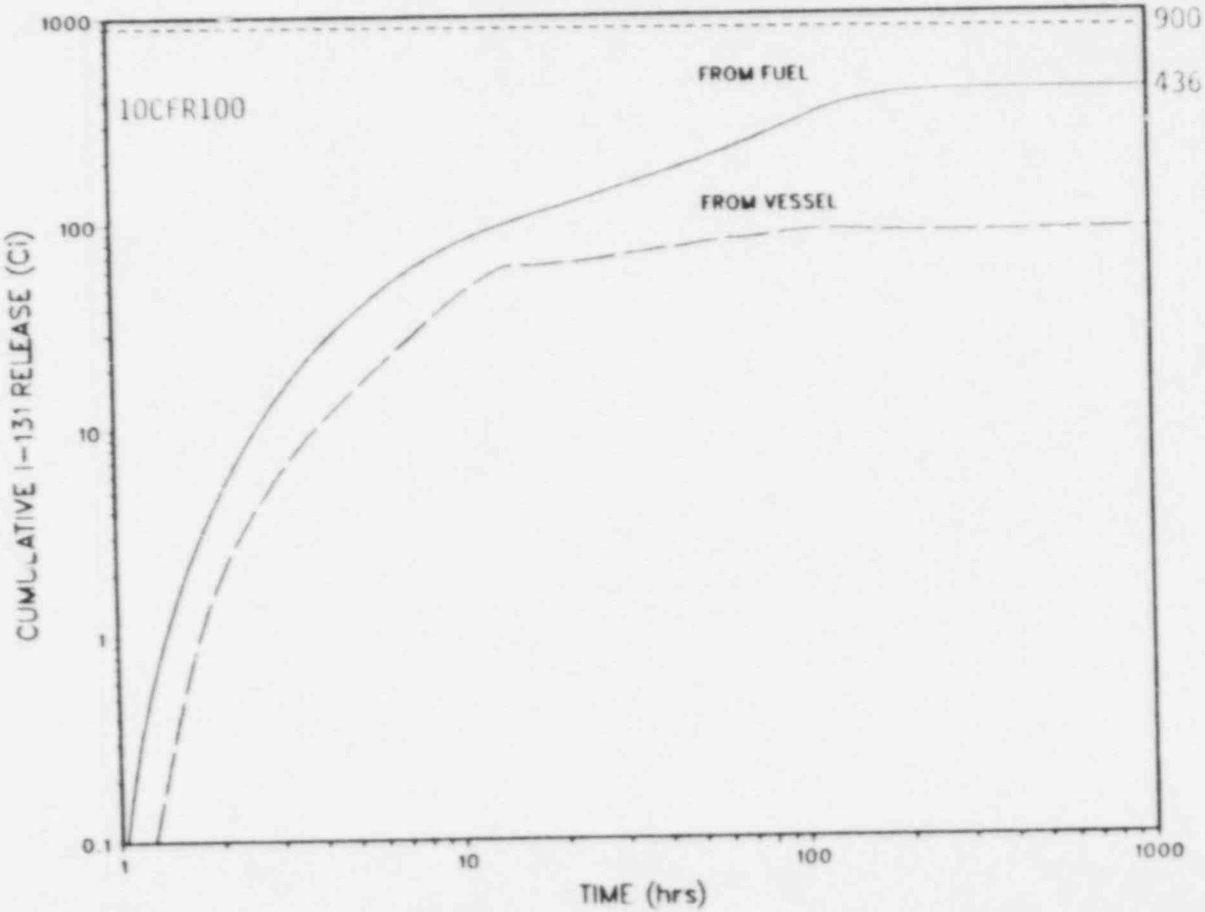
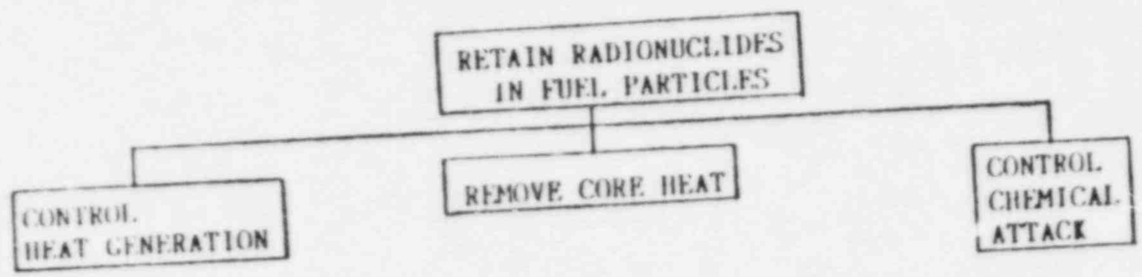
CUMULATIVE I–131 RELEASE FOR THE DEPRESSURIZED CONDUCTION COOLDOWN



DEVELOPMENT OF NEXT TIER OF FUNCTIONS BELOW "RETAIN RADIONUCLIDES IN FUEL PARTICLES"



EG&G Idaho, Inc.

EQUIPMENT CLASSIFICATION

- DEFINITIONS
- METHOD / APPLICATION

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EQUIPMENT CLASSIFICATION DEFINITION

SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS (SSCs) ARE THOSE PERFORMING REQUIRED FUNCTIONS TO MEET 10CFR100 DOSES FOR DBEs

SAFETY RELATED SSCs ARE DESCRIBED AND THEIR LIMITING DESIGN CONDITIONS EVALUATED IN THE SARs (PSID)

SELECTION OF SAFETY-RELATED SYSTEMS STRUCTURES AND COMPONENTS

STEP 1:

FOR EACH DBE, CLASSIFY AS SAFETY–RELATED THOSE SSCs, NEEDED FOR COMPLIANCE WITH 10CFR100 DOSE CRITERIA

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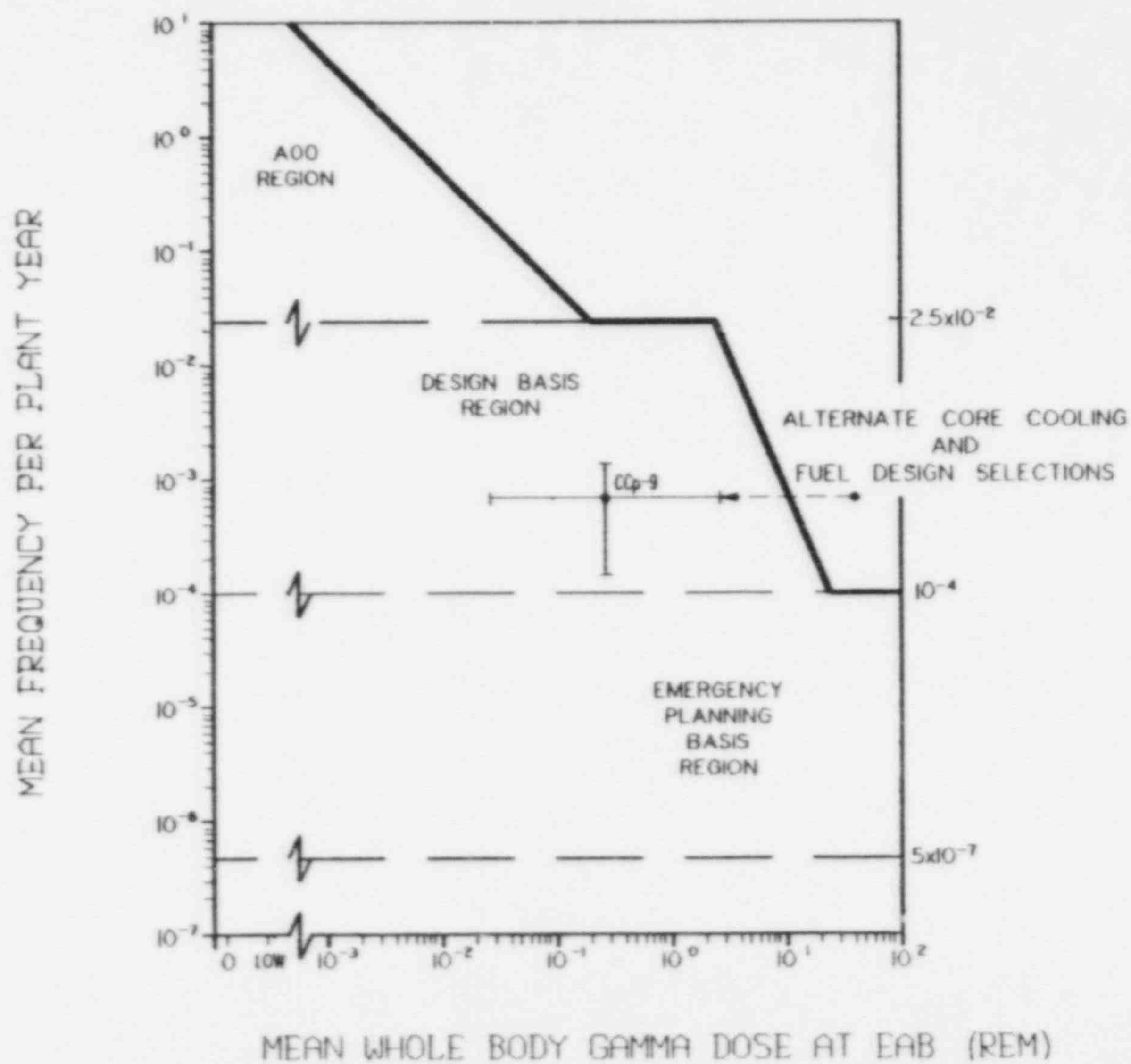
EXAMPLE SELECTION OF SAFETY-RELATED SSC



SSC2

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SELECTION OF SAFETY-RELATED SYSTEMS STRUCTURES AND COMPONENTS

STEP 2:

FOR EACH EPBE WITH CONSEQUENCES GREATER THAN 10CFR100 CLASSIFY AS SAFETY-RELATED THOSE SSC DESIGN SELECTIONS NEEDED TO ASSURE THAT THE EVENT FREQUENCY IS BELOW THE DESIGN BASIS REGION

STEP 3:

FOR EACH SSC CLASSIFIED AS SAFETY RELATED, DETERMINE THE LIMITING DESIGN CONDITIONS FOR IT'S OPERATION BY EXAMINING ALL IT'S ASSOCIATED DBEs

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SAFETY RELATED SSCs NEEDED FOR RADIONUCLIDE RETENTION WITHIN FUEL

FUEL QUALITY

- AS–MANUFACTURED:
  - CONTAMINATION FRACTION
  - DEFECTIVE SiC COATING FRACTION
- NORMAL OPERATION AND TRANSIENTS
  - COATING FAILURE FRACTION

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SAFETY RELATED SSCs NEEDED TO REMOVE CORE HEAT

CORE POWER DENSITY	5.9 W/CC
CORE GEOMETRY AND DIMENSIONS	ANNULAR 1.65 M ID 3.50 M OD
CORE HIGH TEMPERATURE MATERIAL PROPERTIES	GRAPHITE
VESSEL MATERIAL PROPERTIES	CARBON
HEAT SINK	REACTOR CAVITY COOLING SYSTEM

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REPRESENTATIVE CLASSIFICATION OF AN SSC AS SAFETY-RELATED FOR THE MHTGR

REQUIRED FUNCTION: REMOVE CORE HEAT

SSC AVAILABLE TO PERFORM FUNCTION ?	DBE-1 PC-4	DBE-5 PC-9	DBE-8 CCL-N1	DBE-11 CCP-9	DBE-14 PC-5	CLASS.
1. MAIN LOOP	YES				YES	
2. SHUTDOWN COOLING SYSTEM	YES	YES			YES	
3. REACTOR CAVITY COOLING (ACTIVE MODE)	YES	YES		YES	YES	
4. REACTOR CAVITY COOLING (PASSIVE MODE)	YES	YES	YES	YES	YES	X
5. REACTOR CAVITY & SURROUNDINGS	YES	YES		YES	YES	

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LICENSING BASIS SUMMARY

- METHOD USES FUNCTIONAL ANALYSIS (IA), PRA AND RESULTING DESIGN SELECTIONS TO SHOW COMPLIANCE WITH TOP LEVEL REGULATION CRITERIA
- METHOD PROVIDES SYSTEMATIC, TRACEABLE PROCESS TO DERIVE LICENSING BASES SPECIFIC TO MHTGR
- APPLICATION OF METHOD DEMONSTRATES MHTGR EMPHASIS ON RADIATION RETENTION WITHIN FUEL WITH PASSIVE FEATURES
- APPROACH IS CONSISTENT WITH DRAFT ADVANCED REACTOR POLICY

NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

DESIGN OVERVIEW

A. J. NEYLAN
DIVISION DIRECTOR

GA TECHNOLOGIES

MAJOR NSSS SYSTEMS

- REACTOR SYSTEM
- HEAT TRANSPORT SYSTEM
- VESSEL SYSTEM
- SHUTDOWN COOLING SYSTEM
- REACTOR CAVITY COOLING SYSTEM (IN BOP)
- PLANT CONTROL & PROTECTION SYSTEMS
- FUEL HANDLING SYSTEM
- REACTOR SERVICE SYSTEMS

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NUCLEAR STEAM SUPPLY SYSTEM

CONFIGURATION

DESIGN PARAMETERS

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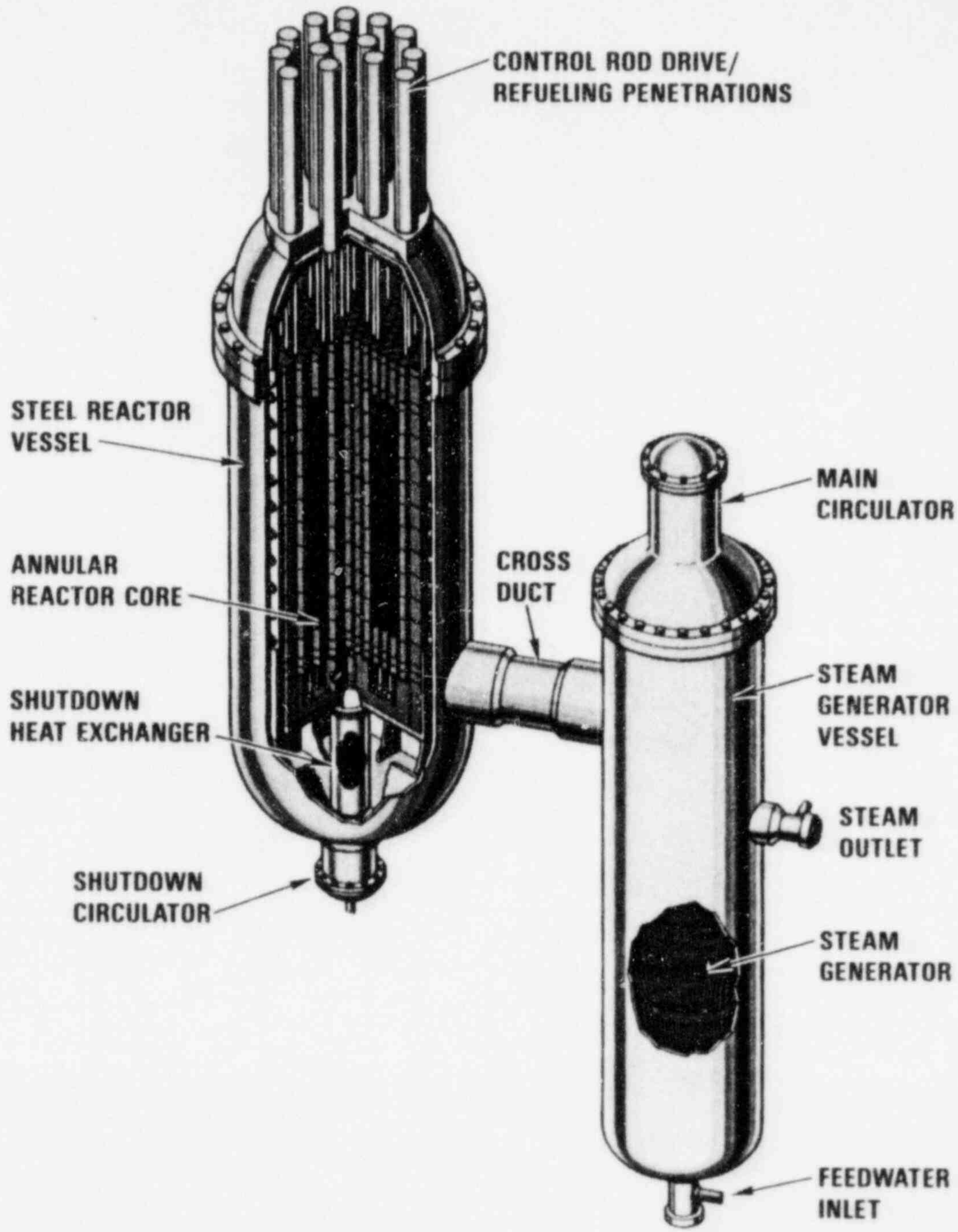
CONFIGURATION

- SEPARATE REACTOR & STEAM GENERATOR STEEL VESSELS
- CONCENTRIC CROSS DUCT
- PRISMATIC FUEL - DOWNFLOW ANNULAR CORE
- CORE BARREL SUPPORT STRUCTURE
- STEAM GENERATOR - ONCE THRU/UPHILL BOILING
- TOP-MOUNTED MAIN CIRCULATOR - IN COLD LEG OF LOOP
- ELECTRIC MOTOR DRIVE MAIN CIRCULATOR
- SEPARATE SHUTDOWN COOLING SYSTEM - HEAT EXCHANGER AND CIRCULATOR BELOW CORE
- LOCATED IN SILO IN BOP-NUCLEAR ISLAND

350 MW(t) MODULAR HTGR PLANT



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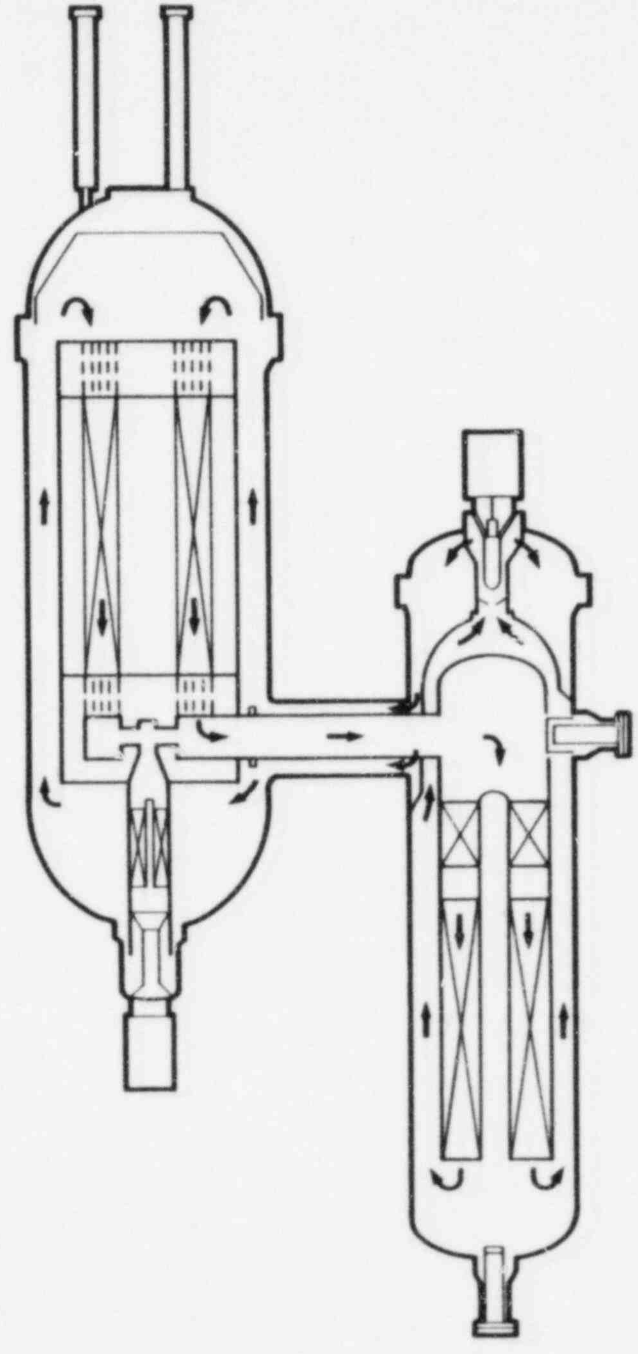
350 MW(t) MODULAR HTGR PLANT HELIUM FLOW NORMAL OPERATION



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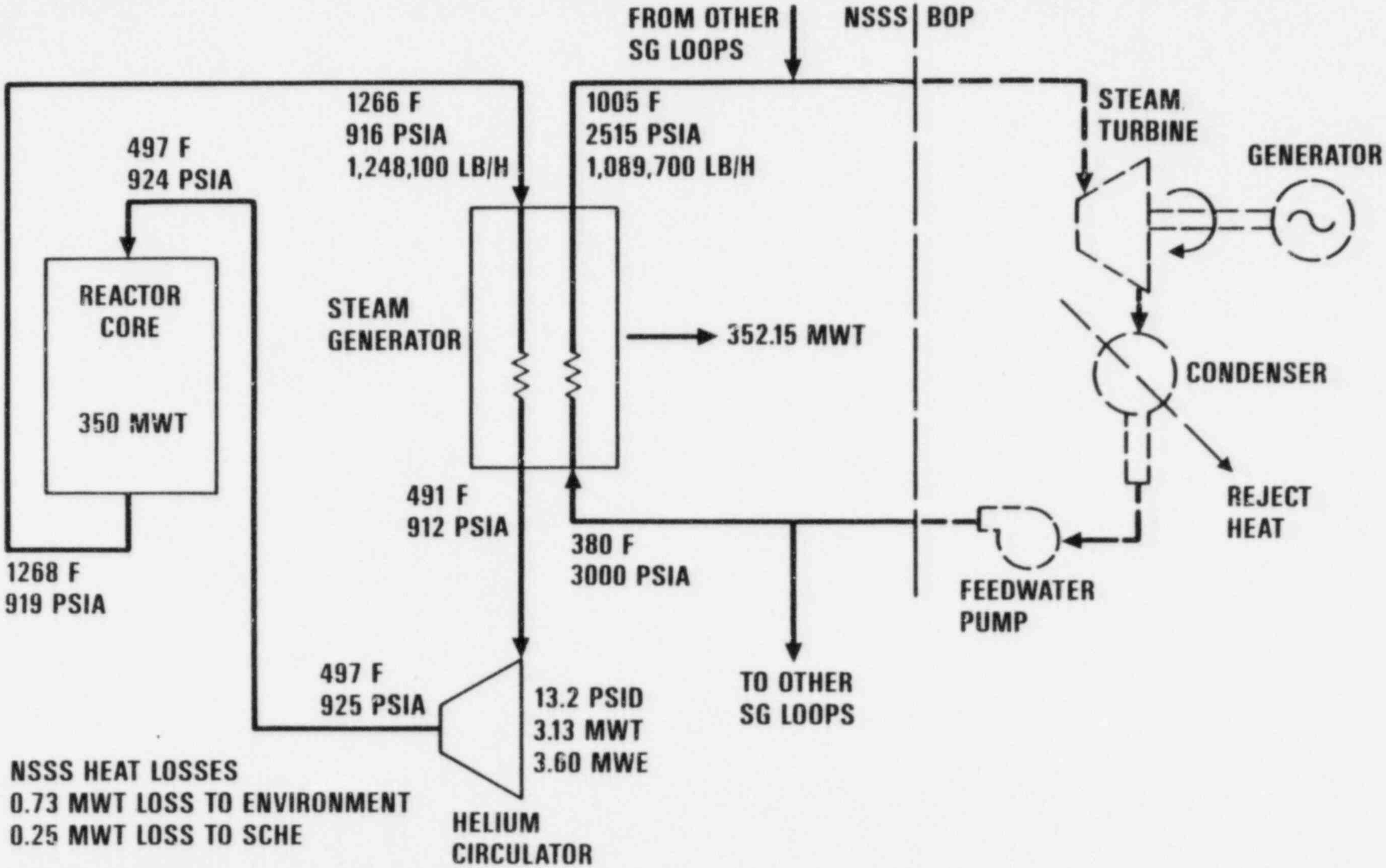
NSSS HEAT BALANCE DIAGRAM
4 x 350 MWT MODULAR HTGR PLANT



NSSS HEAT LOSSES
0.73 MWT LOSS TO ENVIRONMENT
0.25 MWT LOSS TO SCHE

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DESIGN PARAMETERS

o	CORE POWER RATING (4 MDLS)	1400 MW(t)
o	CORE POWER DENSITY	5.91 W/cm^3
o	HELIUM TEMP. CORE INLET	497°F
o	HELIUM TEMP. CORE OUTLET	1268°F
o	HELIUM PRESSURE	925 psia
o	CIRCULATOR SPEED/HP	5,720 rpm/4,800
o	FEEDWATER PRESSURE	3,000 psia
o	FEEDWATER TEMP.	380°F
o	STEAM TEMP.	1005°F
o	STEAM PRESSURE	2515 psig
c	STEAM GENERATOR OUTPUT	352 MW(t)

350 MW(t) MODULAR REACTOR CORE CROSS SECTION

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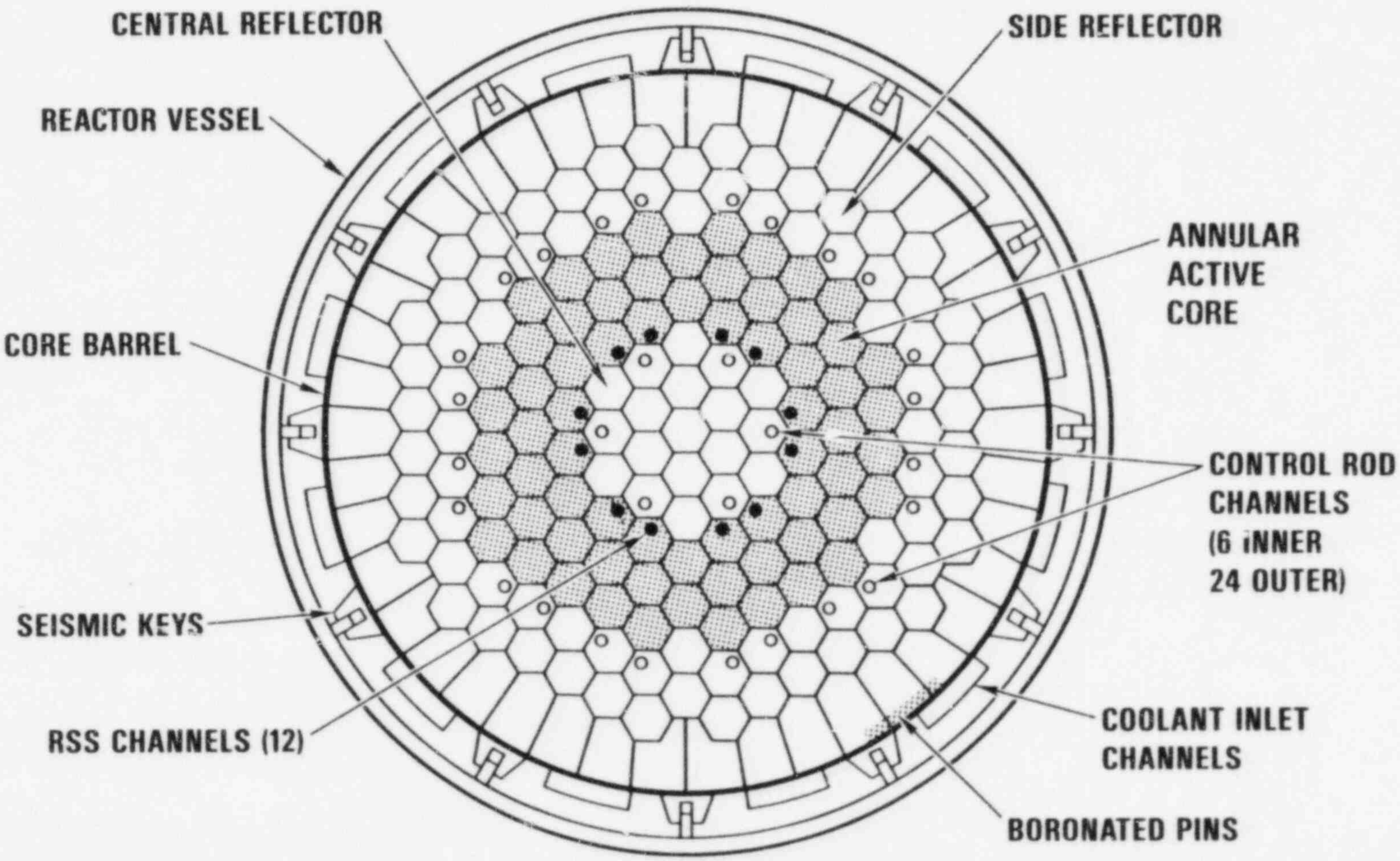

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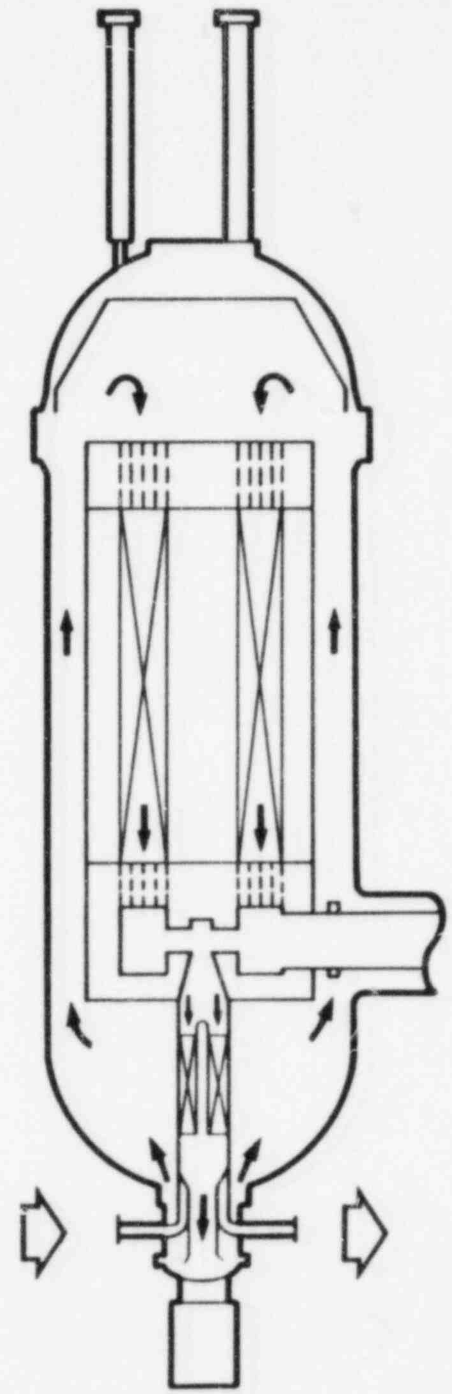
DECAY HEAT REMOVAL

HEAT TRANSPORT SYSTEM (HTS)

SHUTDOWN COOLING SYSTEM (SCS)

REACTOR CAVITY COOLING SYSTEM (RCCS)

DECAY HEAT REMOVAL WITH SHUTDOWN COOLING SYSTEM



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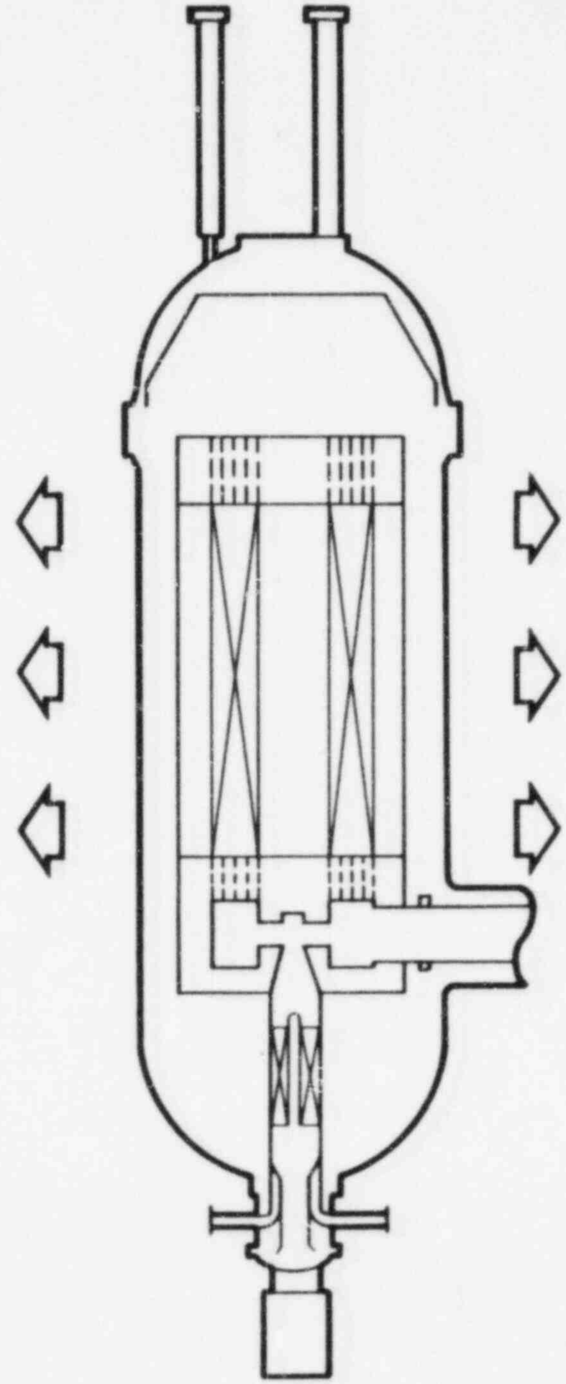
6-28-85

DECAY HEAT REMOVAL BY CONDUCTION AND RADIATION



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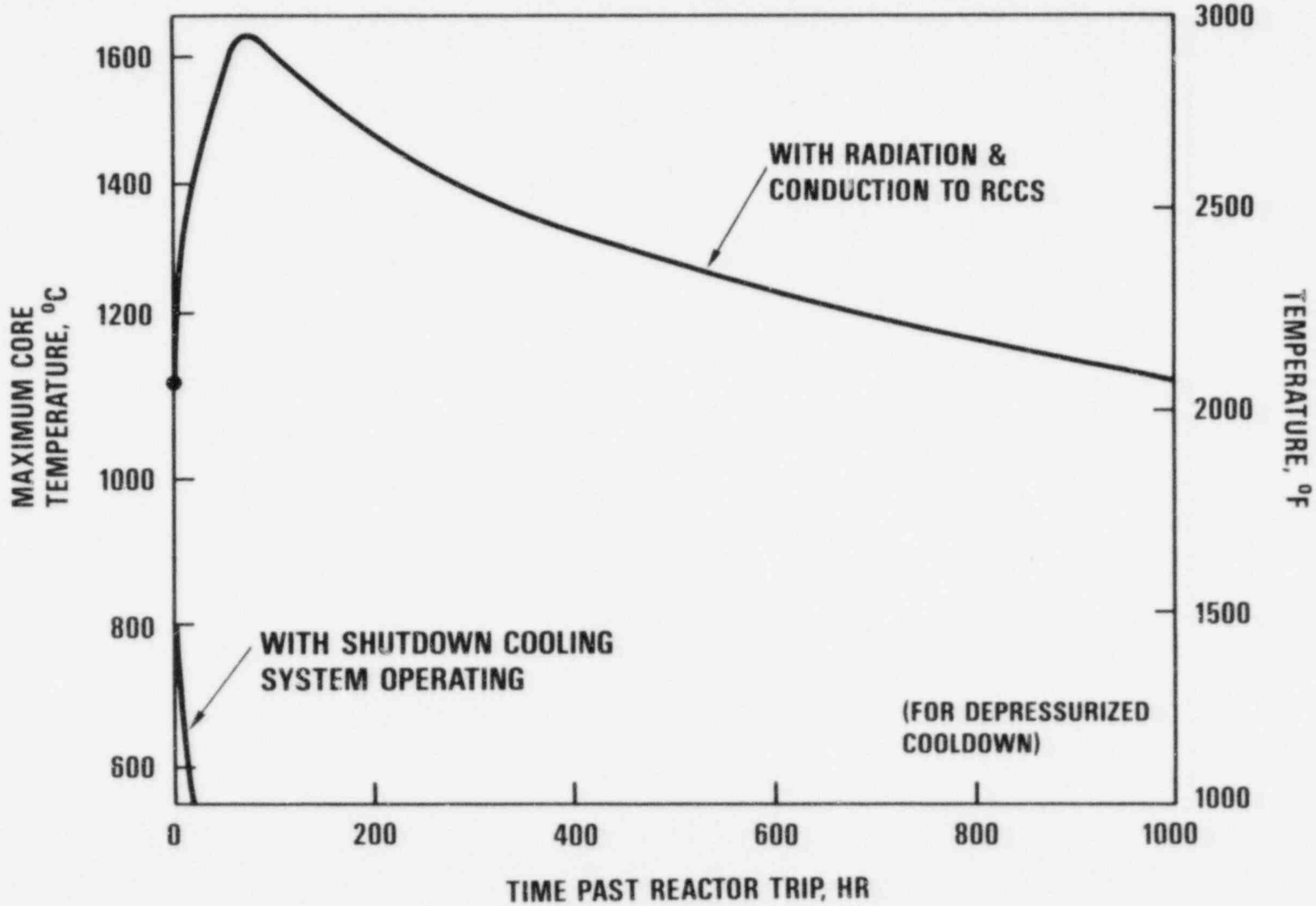
7-8-85

CORE TEMPERATURES REMAIN BELOW DESIGN LIMITS DURING ACCIDENTS



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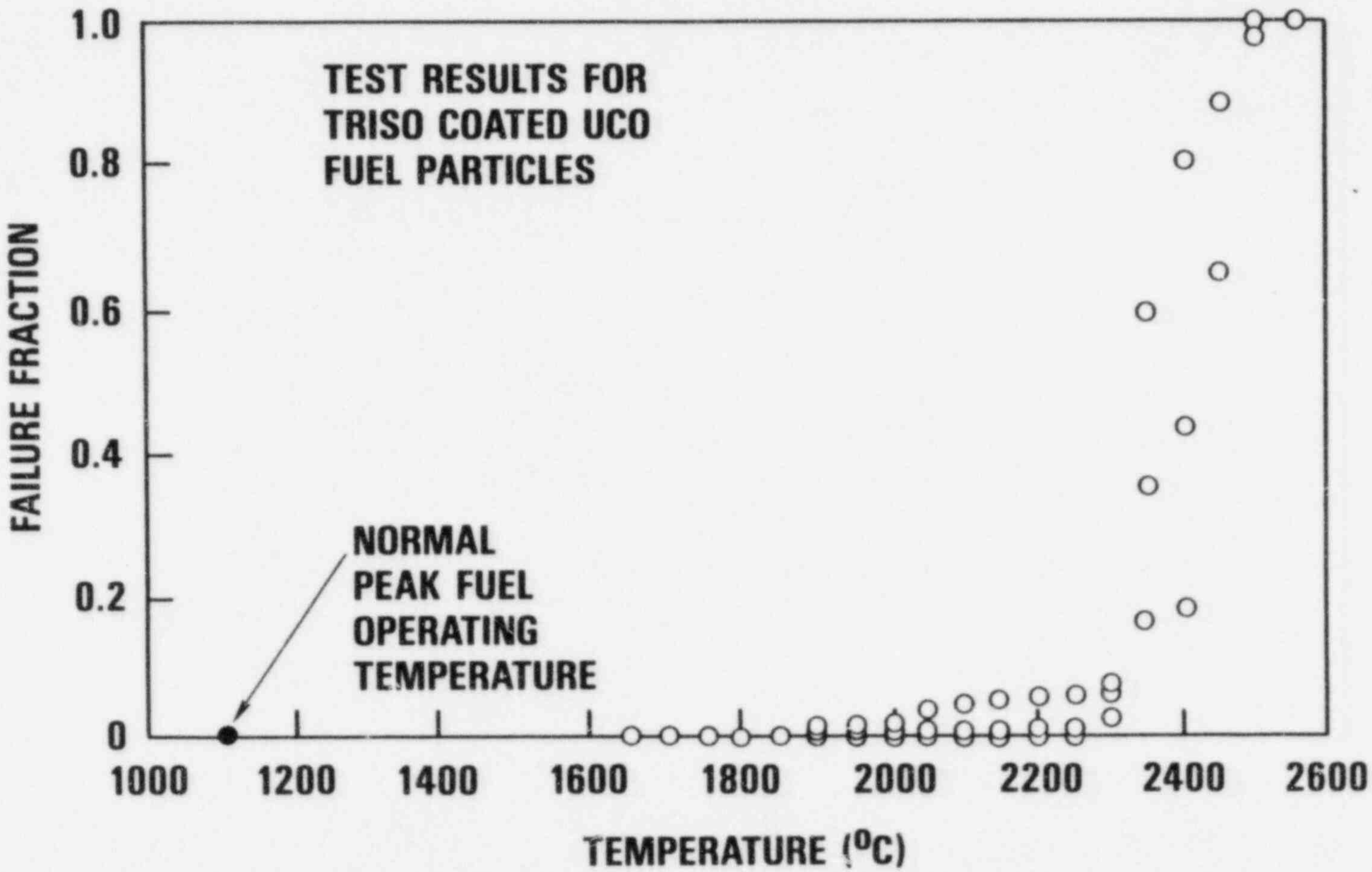
12-6-85

COATED FUEL PARTICLES MAINTAIN INTEGRITY UP TO 1800°C



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9-4-84

ACRS BRIEFING OBJECTIVES

- BRIEF ACRS ON APPROACH TO DESIGN
- BRIEF ACRS ON LICENSING APPROACH AND METHODOLOGY WHICH HAS BEEN PROPOSED TO NRC STAFF
- BRIEF ACRS ON MHTGR DESIGN STATUS
- RECEIVE ACRS COMMENTS ON PROPOSED LICENSING APPROACH AND METHODOLOGY

PROGRAM OBJECTIVE

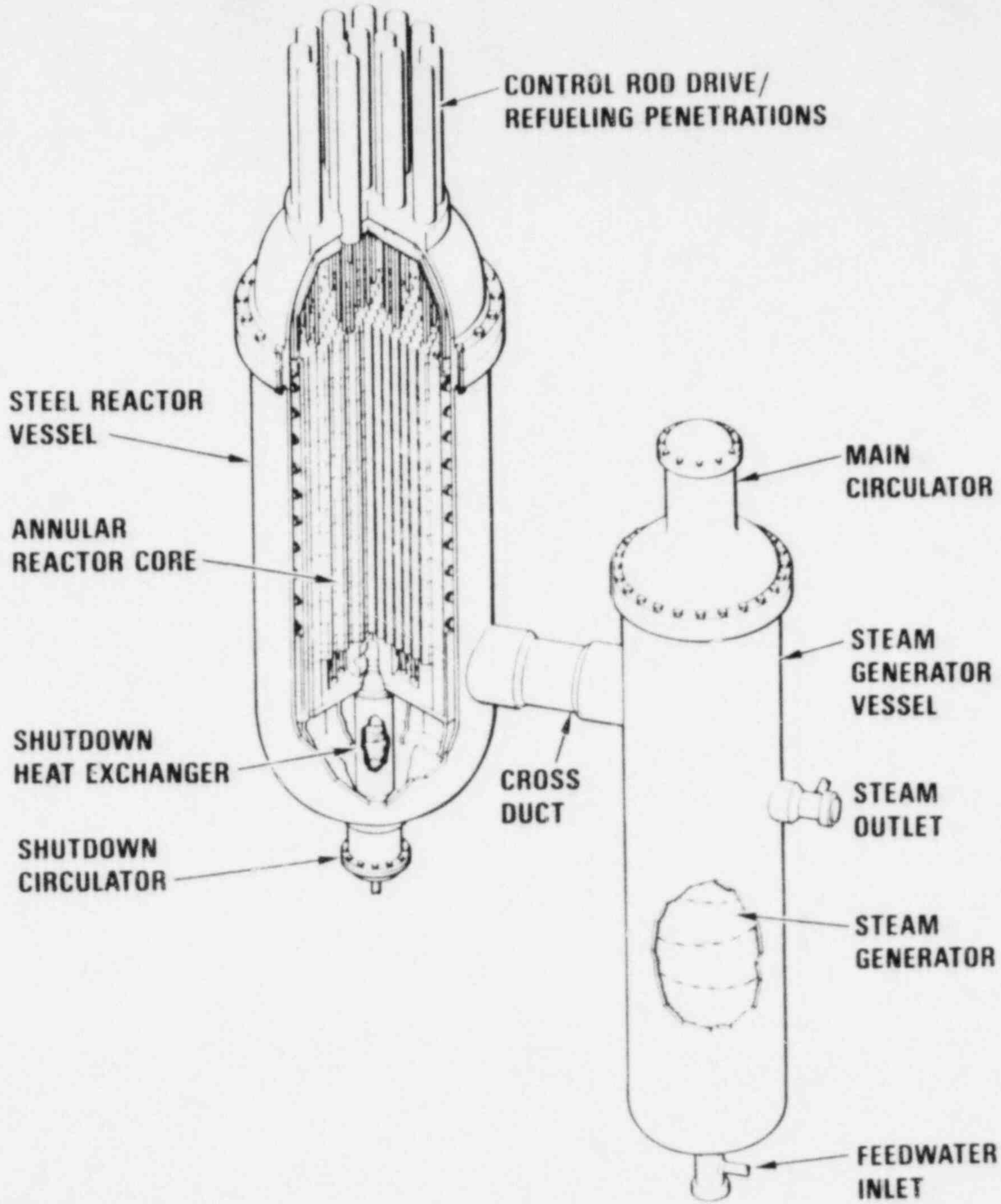
DEVELOP HTGR's FOR BROAD RANGE OF APPLICATIONS IN SUPPORT OF COMMERCIAL/USER INTERESTS IN SAFETY AND HIGHER TEMPERATURE CHARACTERISTICS OF THESE PLANTS.

HTGR PROGRAM PARTICIPANTS

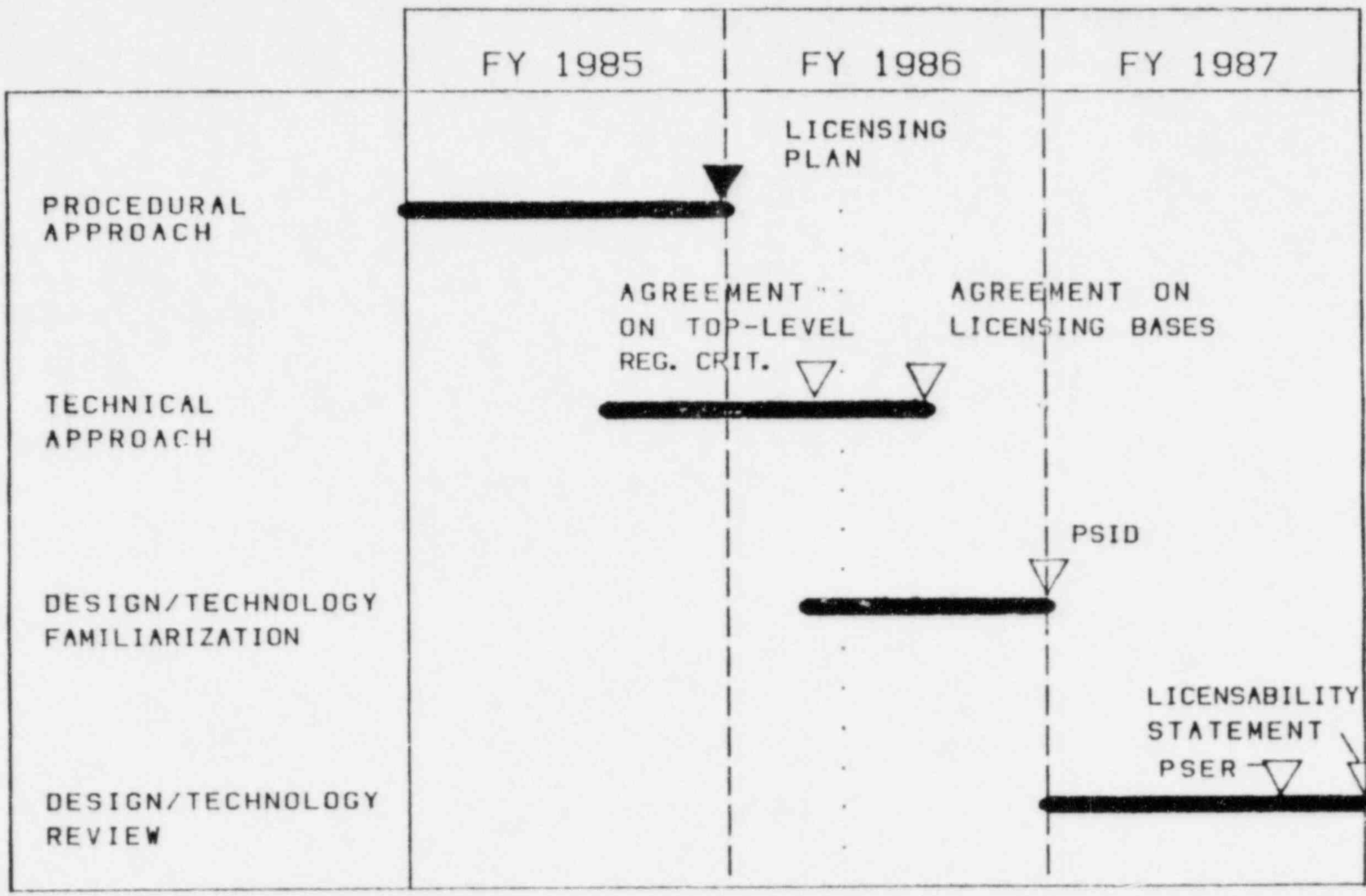
- GA TECHNOLOGIES INC.
- GENERAL ELECTRIC COMPANY
- COMBUSTION ENGINEERING, INC.
- STONE & WEBSTER ENGINEERING, INC.
- BECHTEL GROUP, INC.
- OAK RIDGE NATIONAL LABORATORY
- IDAHO NATIONAL ENGINEERING LABORATORY
- EG&G IDAHO, INC.
- GAS-COOLED REACTOR ASSOCIATES

MODULAR

HTGR



NRC INTERACTIONS



PROCEDURAL APPROACH INTERACTION

LICENSING PLAN DRAFT	COMPLETE
NRC ADVANCED REACTOR POLICY ISSUED FOR COMMENT	COMPLETE
LICENSING PLAN FORMAL SUBMITTAL	COMPLETE
INDUSTRY COMMENTS ON POLICY	COMPLETE
NRC ACCEPTANCE OF LICENSING PLAN	COMPLETE

TECHNICAL APPROACH INTERACTION

	BRIEFING	SUBMITTAL
TOP-LEVEL CRITERIA	COMPLETE	COMPLETE
BRIDGING METHODS	COMPLETE	2/86
ACCIDENT SELECTION METHOD	COMPLETE	2/86
SAFETY CLASS SELECTION	COMPLETE	2/86
PRINCIPAL DESIGN CRITERIA	COMPLETE	2/86
ACRS BRIEFING	1/86	NA

DESIGN/TECHNOLOGY FAMILIARIZATION

ISSUE	BRIEFING
MODULAR HTGR DESIGN	12/85
FUEL	3/86
DECAY HEAT REMOVAL	5/86
REACTIVITY CONTROL	5/86
CORE SUPPORT STRUCTURE	5/86
ISI	7/86
WATER/AIR INGRESS	7/86
CONTAINMENT/CONFINEMENT	8/86
BOP CLASSIFICATION	8/86
MULTIPLE MODULE CONTROL	8/86
STANDARD PLANT ISSUES	8/86
ACRS BRIEFING	9/86

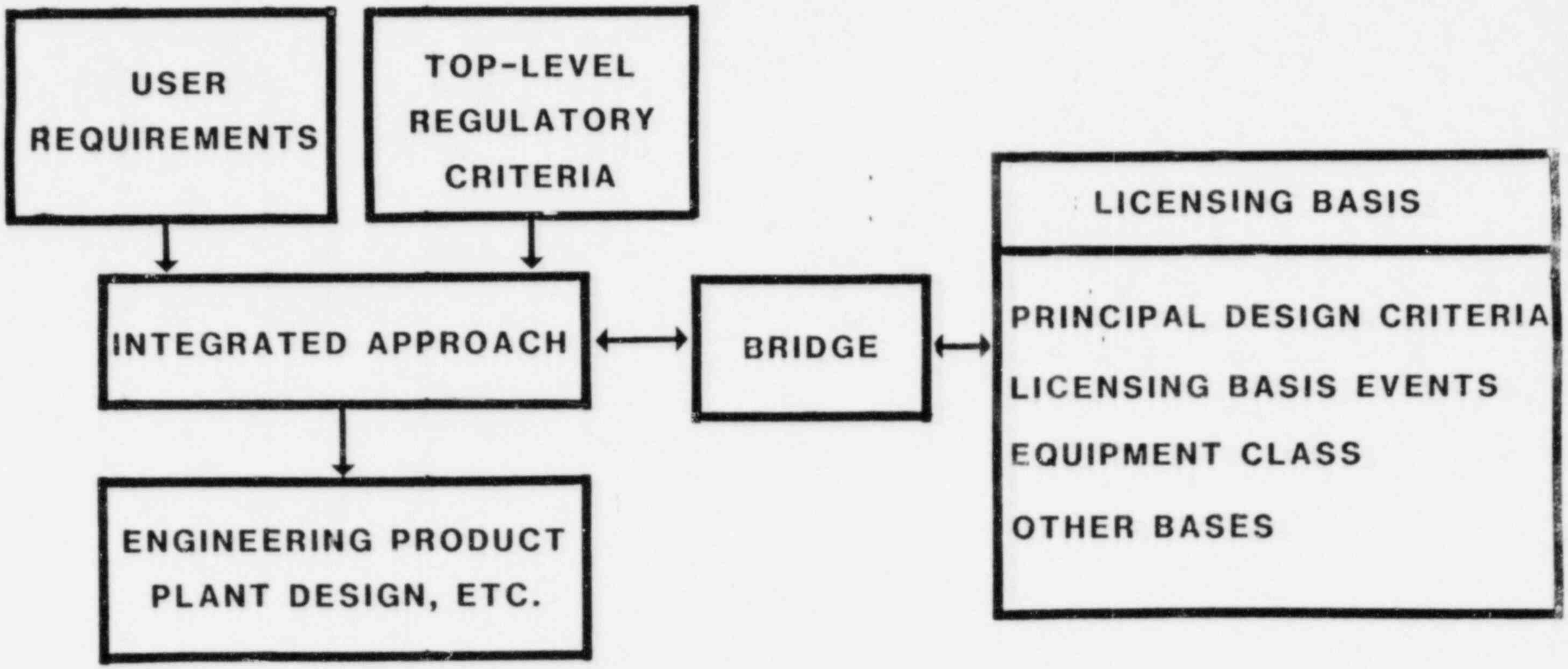
DESIGN & TECHNOLOGY FAMILIARIZATION(cont)

DOCUMENT	DATE
TECHNOLOGY PLAN	9/86
PRA	9/86
PSID	
- OUTLINE	COMPLETE
- FULL SUBMITTAL	9/86

DESIGN & TECHNOLOGY REVIEW

PSER	6/87
LICENSABILITY STATEMENT	9/87

HTGR DESIGN & LICENSING APPROACH



SAFETY PHILOSOPHY

- PROVIDE DEFENSE-IN-DEPTH THROUGH PURSUIT OF 4 GOALS:

  1 - MAINTAIN SAFE PLANT OPERATION

  2 - MAINTAIN PLANT PROTECTION

  3 - MAINTAIN CONTROL OF RADIONUCLIDE RELEASE

  4 - MAINTAIN EMERGENCY PREPAREDNESS

- GOAL 1 TO BE ACHIEVED BY HIGHLY RELIABLE OPERATION AND WITH WELL TRAINED PERSONNEL

- GOALS 2 & 3 TO BE ACHIEVED THROUGH UTILIZATION OF INHERENT CHARACTERISTICS AND PASSIVE SAFETY FEATURES

- GOALS 1 - 3 TO BE ACHIEVED SO WELL THAT MINIMAL RELIANCE NEED BE PLACED ON GOAL 4

LICENSING APPROACH SUMMARY

THREE STEPS:

1) IDENTIFY TOP-LEVEL CRITERIA GENERIC TO ALL REACTOR TYPES AS STARTING POINT.

   - DONE.

2) DEVELOP PROCESS TO DERIVE LICENSING BASES SPECIFIC TO THE MHTGR WHICH ENSURE THAT TOP-LEVEL CRITERIA ARE MET.

   - DONE.

3) APPLY PROCESS TO IDENTIFY MHTGR LICENSING BASES.

   - IN PROGRESS.

PURPOSE OF THE INTEGRATED APPROACH

THE INTEGRATED APPROACH IS USED TO:

- o DEVELOP REQUIREMENTS
- o EVALUATE DESIGNS SELECTED TO MEET REQUIREMENTS
- o COMMUNICATE

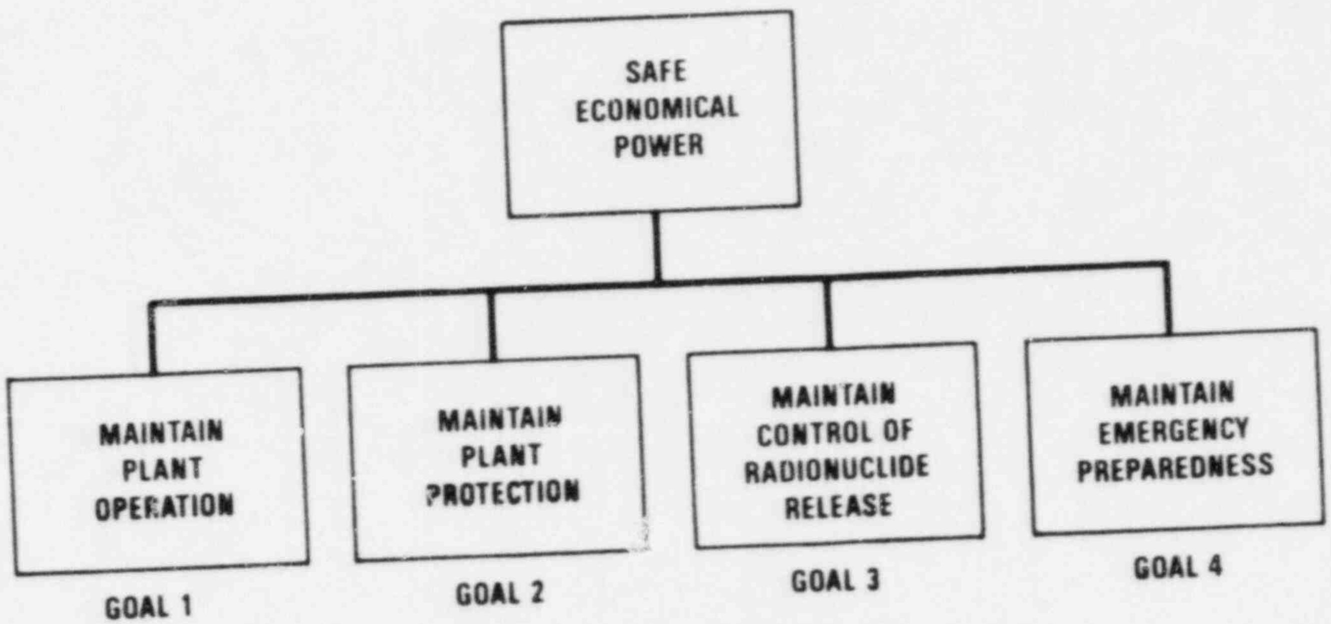
SAVINGS FROM THE INTEGRATED APPROACH

THE SAVINGS ENVISIONED FROM THE USE OF THE INTEGRATED APPROACH ARE DUE TO:

- A CLEAR UNDERSTANDING BY THE DESIGNERS, CONTRACTORS AND OPERATORS OF WHAT THEIR ROLES AND RESPONSIBILITIES ARE.
- AN EARLY IDENTIFICATION OF INTERFACES WHICH REDUCES THE RISK OF LATER MORE COSTLY REVISIONS.
- VISIBILITY OF THE BASIS FOR DESIGN REQUIREMENTS.
- ELIMINATION OF UNJUSTIFIABLE RETROFITS.
- JUSTIFICATION FOR, OR DELETION OF, DEVELOPMENT PROGRAMS.

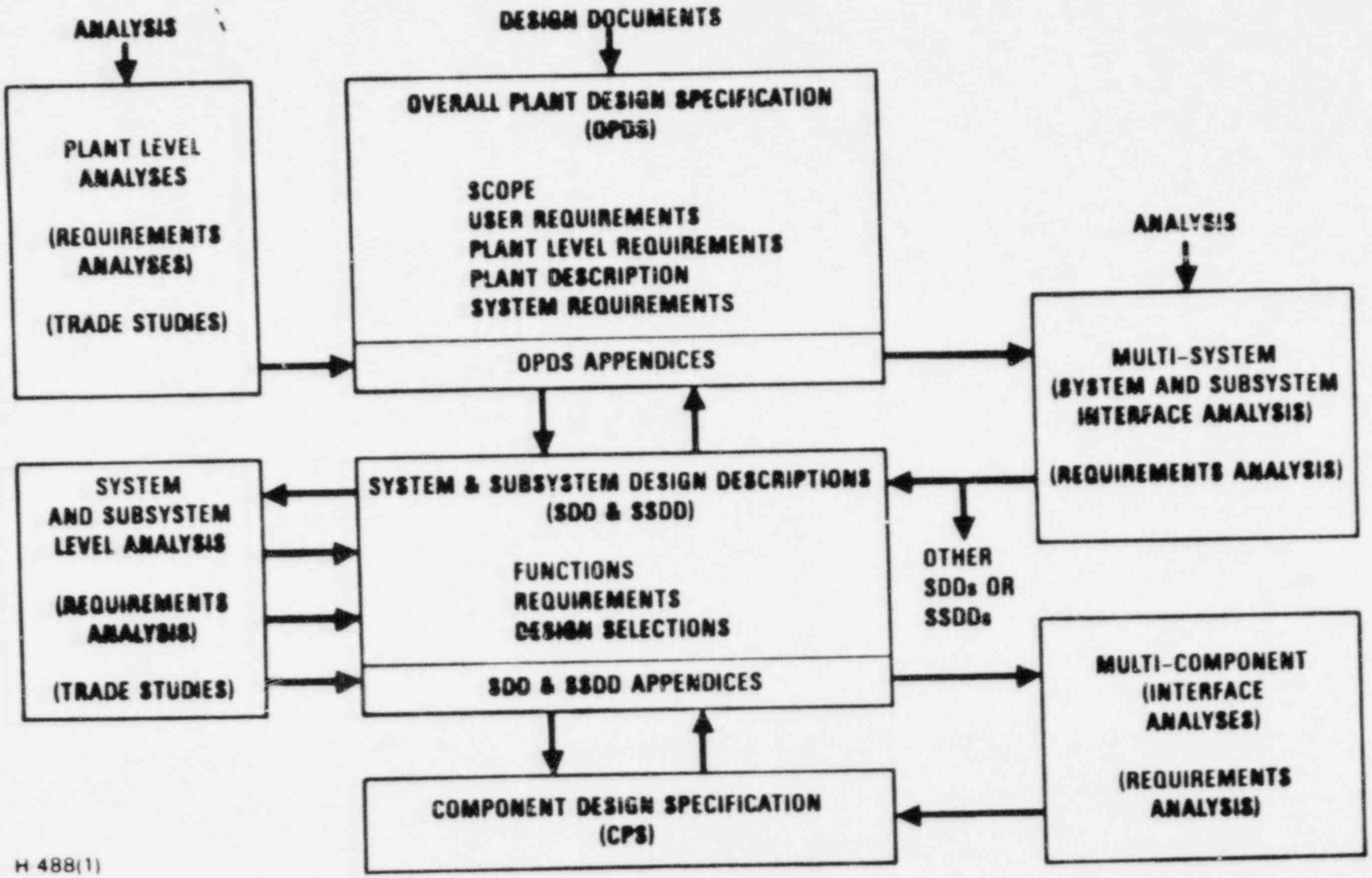
G-817(6)
8-13-84

PLANT GOALS

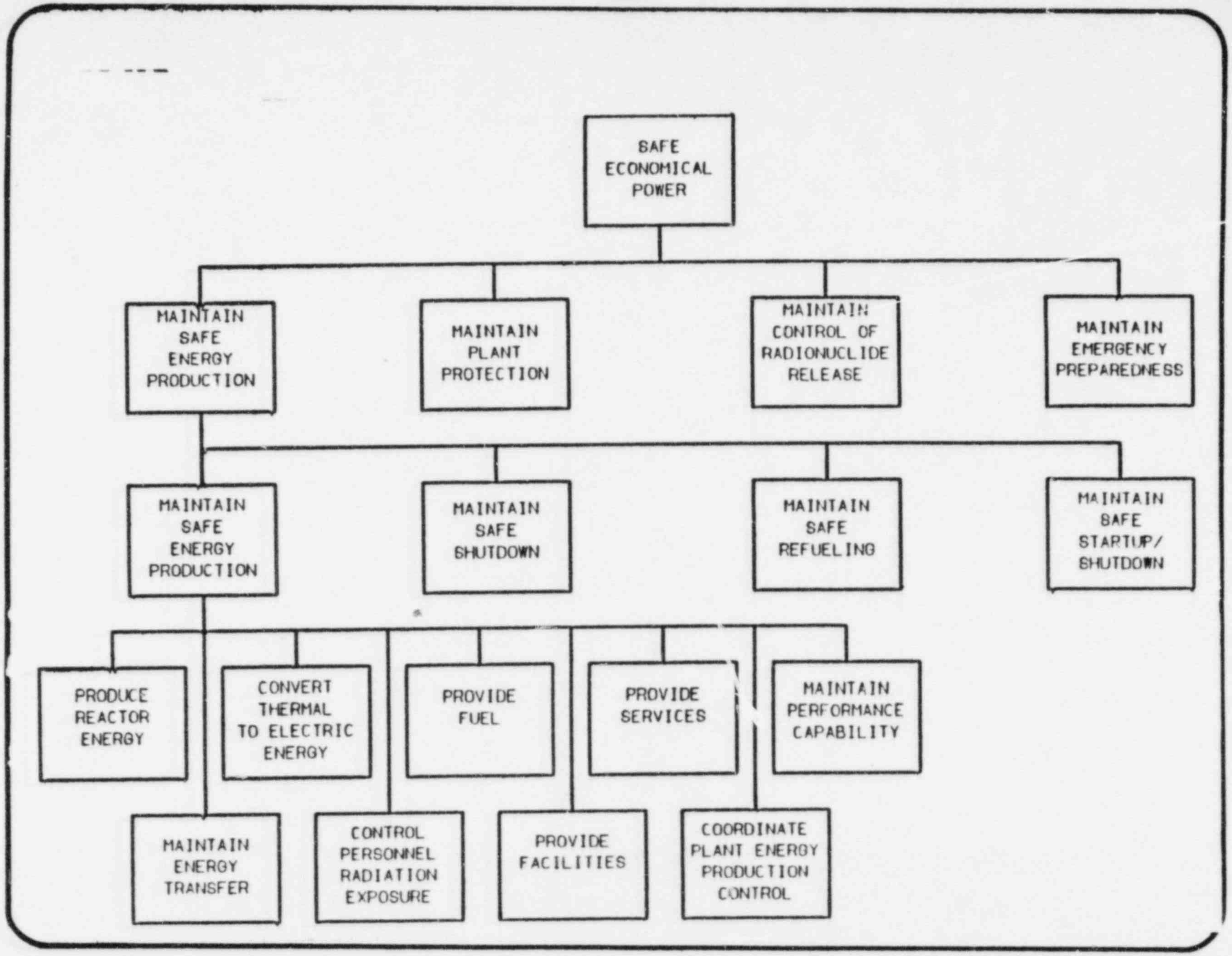


G-817(16)
8-13-84

DOCUMENTATION DEVELOPMENT LOGIC



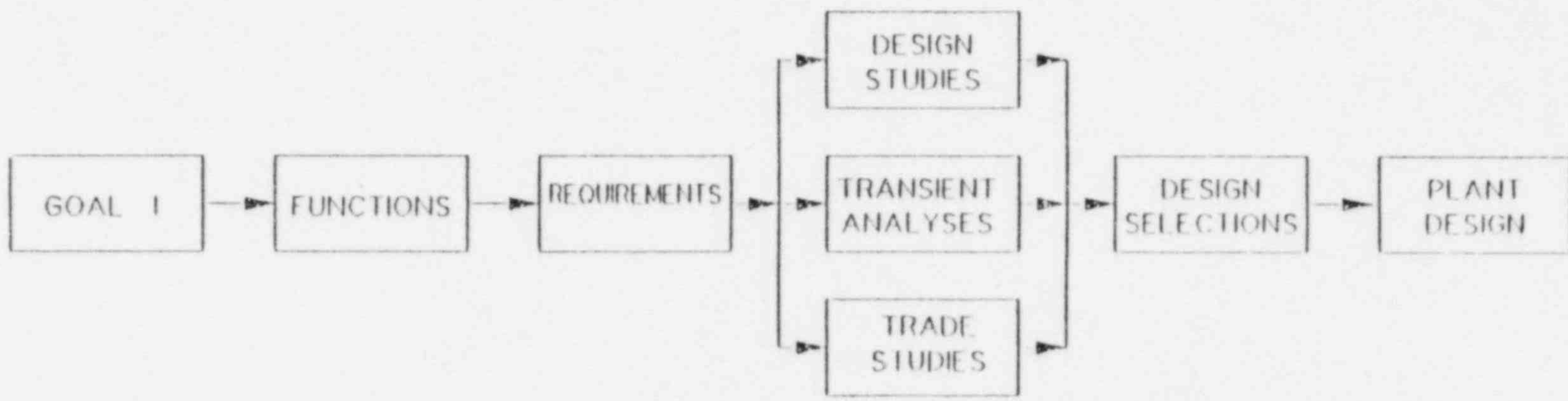
H 488(1)
12-10-85


SAFE ECONOMICAL POWER
MAINTAIN SAFE ENERGY PRODUCTION
MAINTAIN PLANT PROTECTION
MAINTAIN CONTROL OF RADIONUCLIDE RELEASE
MAINTAIN EMERGENCY PREPAREDNESS
MAINTAIN SAFE ENERGY PRODUCTION
MAINTAIN SAFE SHUTDOWN
MAINTAIN SAFE REFUELING
MAINTAIN SAFE STARTUP/ SHUTDOWN
PRODUCE REACTOR ENERGY
CONVERT THERMAL TO ELECTRIC ENERGY
PROVIDE FUEL
PROVIDE SERVICES
MAINTAIN PERFORMANCE CAPABILITY
MAINTAIN ENERGY TRANSFER
CONTROL PERSONNEL RADIATION EXPOSURE
PROVIDE FACILITIES
COORDINATE PLANT ENERGY PRODUCTION CONTROL

INTEGRATED APPROACH TO POWER PLANT DESIGN GOAL 1 DESIGN:

- DEVELOP FUNCTIONS AND REQUIREMENTS AND MAKE DESIGN SELECTIONS TO MEET GOAL 1 REQUIREMENTS TO MAINTAIN NORMAL PLANT OPERATION.

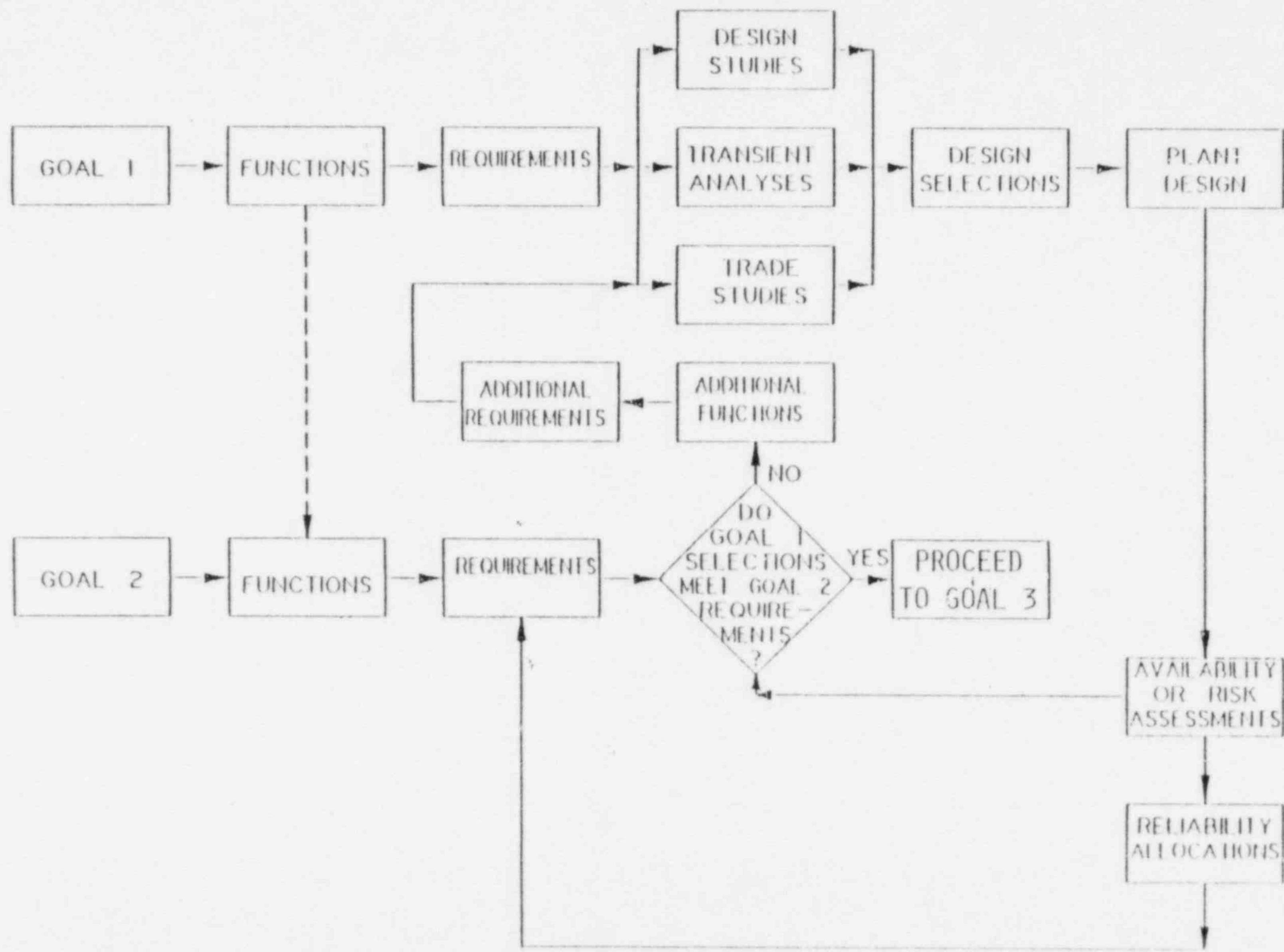
INTEGRATED APPROACH TO DESIGN

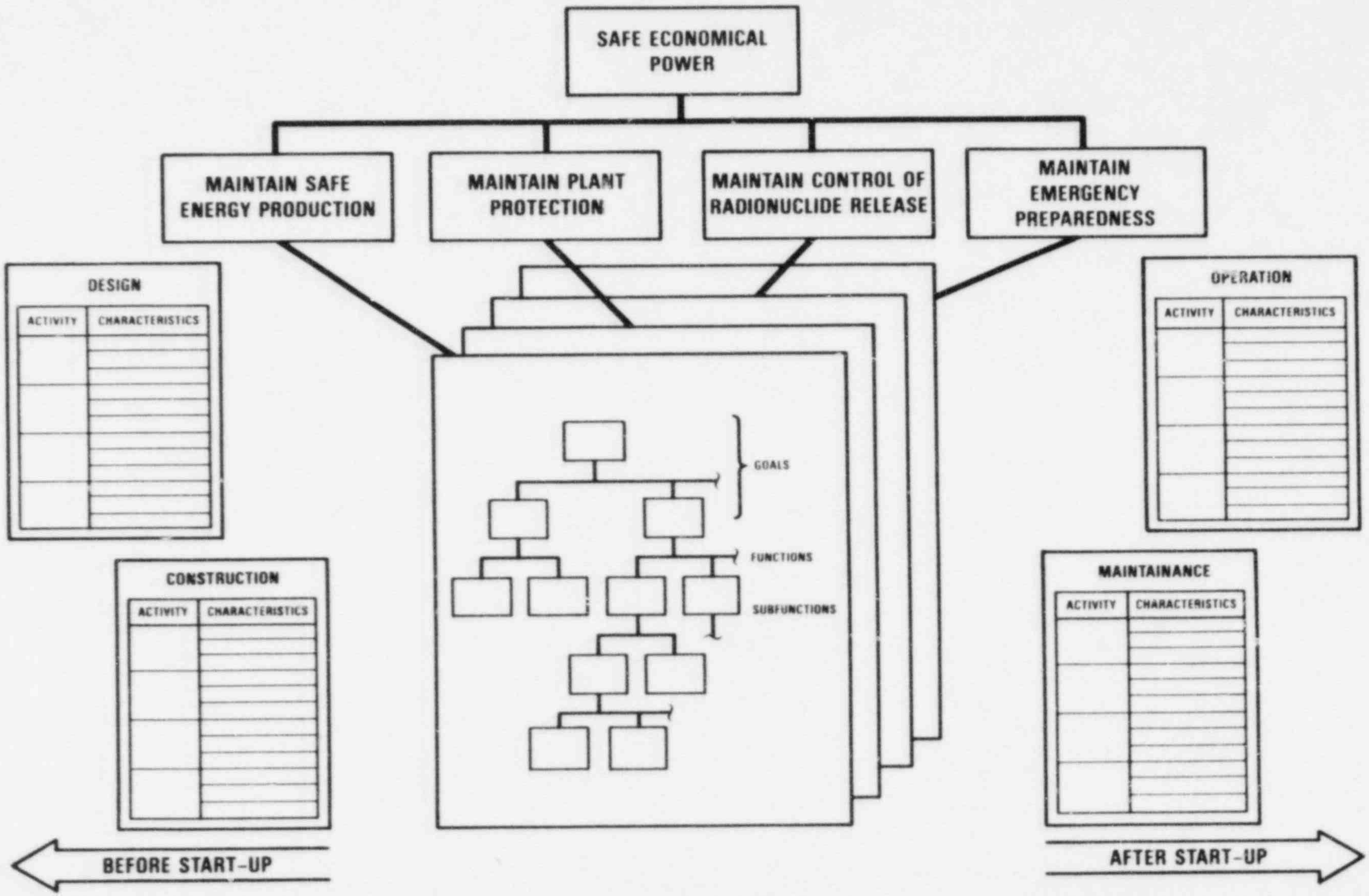


INTEGRATED APPROACH TO POWER PLANT DESIGN GOAL 1-2 DESIGN:

- START WITH GOAL 1 DESIGN SELECTIONS
- DEVELOP GOAL 2 FUNCTIONS AND REQUIREMENTS TO MAINTAIN PLANT PROTECTION
- DETERMINE IF GOAL 1 DESIGN SELECTIONS MEET THE GOAL 2 FUNCTIONS AND REQUIREMENTS
- IF REQUIRED MODIFY OR SUPPLEMENT GOAL 1 DESIGN SELECTIONS

INTEGRATED APPROACH TO DESIGN





H-488(2)
12-10-85

HTGR BRIEFING
FOR THE ACRS SUBCOMMITTEE
ON ADVANCED REACTORS
JANUARY 30, 1986

TOP-LEVEL REQUIREMENTS/CRITERIA FOR HTGRs

PRESENTATION BY

ARCHIE P. KELLEY, JR.
GAS-COOLED REACTOR ASSOCIATES

TOP-LEVEL REGUIREMENTS/CRITERIA FOR HTGRs

- PURPOSE
- UTILITY/USER REQUIREMENTS
- REGULATORY CRITERIA
  - BASIS FOR SELECTION
  - PROPOSED CRITERIA

TOP-LEVEL USER CRITERIA

OVERALL - SAFE, ECONOMICAL NUCLEAR POWER

- 10% ECONOMIC ADVANTAGE OVER COAL ALTERNATIVE
- SITING ENVELOPE COVERING 85% OF U.S. SITES
- SERVICE LIFE OF 40 YEARS

GOAL 1 - MAINTAIN SAFE PLANT OPERATION

EQUIVALENT UNAVAILABILITY OWING TO PLANNED OUTAGES LESS THAN 10%

TOP-LEVEL USER CRITERIA(cont.)

GOAL 2 - MAINTAIN PLANT PROTECTION

- EQUIVALENT UNAVAILABILITY OWING TO UNPLANNED OUTAGES LESS THAN 10%
- ANNUAL EXPECTED VALUE OF DAMAGE LESS THAN INSURANCE PREMIUM OF $4.5 MILLION
- MEAN LIKELIHOOD OF A LOSS OF A SINGLE REACTOR LESS THAN 10^{-5} PER YEAR

TOP-LEVEL USER CRITERIA(cont.)

GOAL 3 - MAINTAIN CONTROL OF RADIONUCLIDE RELEASE

MEET TOP-LEVEL REGULATORY CRITERIA WITHOUT CREDIT FOR SHELTERING OR EVACUATION OF PUBLIC

- CURRENTLY INTERPRETED AS REQUIRING THAT PAG DOSES BE MET FOR EVENTS WITH MEAN FREQUENCIES GREATER THAN 5×10^{-7} PER YEAR

PROPOSED BASES FOR TOP-LEVEL CRITERIA SELECTION

1) CRITERIA MUST BE DIRECT STATEMENTS OF ACCEPTABLE CONSEQUENCES OR RISKS TO THE PUBLIC OR THE ENVIRONMENT

2) CRITERIA MUST BE INDEPENDENT OF PLANT DESIGN

3) CRITERIA MUST BE QUANTIFIABLE

PROPOSED SOURCES AND CANDIDATES FOR TOP-LEVEL REGULATORY CRITERIA

OVERALL - SAFE, ECONOMICAL POWER

NUREG-0880:

- INDIVIDUAL & SOCIETAL MORTALITY RISKS
- COST BENEFIT INVOKED ONLY IF MORTALITY RISK CRITERIA NOT MET

PROPOSED SOURCES AND CANDIDATES FOR TOP-LEVEL REGULATORY CRITERIA (cont.)

GOAL 1 - MAINTAIN SAFE PLANT OPERATION

- 10CFR20:
  - PERMISSIBLE DOSE LEVELS & ACTIVITY CONCENTRATIONS IN UNRESTRICTED AREAS
- 10CFR50 APPENDIX I:
  - NUMERICAL DOSE GUIDELINES

GOAL 2 - MAINTAIN PLANT PROTECTION

- TO BE COVERED BY OCCUPATIONAL EXPOSURE CRITERIA

PROPOSED SOURCES AND CANDIDATES FOR TOP-LEVEL REGULATORY CRITERIA(cont.)

GOAL 3 - MAINTAIN CONTROL OF RADIONUCLIDE RELEASE

- 10CFR50 APPENDIX I:
  - APPLIED ON AN EXPECTED VALUE BASIS TO EVENTS ANTICIPATED TO OCCUR IN PLANT LIFETIME
- 10CFR100:
  - NUMERICAL DOSE GUIDELINES

GOAL 4 - MAINTAIN EMERGENCY PREPAREDNESS

- EPA-520:
  - PAG DOSES

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: STATUS OF INTERACTIONS ON THE DOE ADVANCED HTGR PROGRAM.

DATE: JANUARY 30, 1986

PRESENTER: THOMAS L. KING

PRESENTER'S TITLE/BRANCH/DIV: SECTION LEADER
SAFETY PROGRAM EVALUATION BRANCH, DIVISION OF SAFETY REVIEW AND OVERSIGHT

PRESENTER'S NRC TEL. NO.: 492-7347

SUBCOMMITTEE: ADVANCED REACTORS

OVERALL REVIEW PLAN

- REVIEW CONCEPTUAL DESIGN AND KEY ISSUES OVER THE NEXT TWO YEARS:

    - AGREE ON CRITERIA

    - ASSESS POTENTIAL OF THE DESIGN FOR SATISFYING THESE CRITERIA

    - ASSESS R&D PROGRAMS SUPPORTING THE DESIGN

    - ISSUE SER AND LICENSABILITY STATEMENT

- BASED ON THE REVIEW OF THE CONCEPTUAL DESIGN IDENTIFY ADDITIONAL STEPS NRC MUST TAKE (INCLUDING RESEARCH) TO BE READY TO PROCESS AN HTGR APPLICATION.

HTGR INTERACTION SCHEDULE

ITEM	BRIEFING TO NRC STAFF	SUBMITTAL TO NRC	NRC ACTION	ACRS BRIEFING
LICENSING APPROACH				
LICENSING PLAN - REV. 2	COMPLETE	2/86	N/A	N/A
TOP LEVEL CRITERIA	COMPLETE	COMPLETE	2/86	1/86
BRIDGING METHODS	COMPLETE	2/86	3/86	1/86
ACCIDENT SELECTION CRITERIA	COMPLETE	2/86	3/86	1/86
LBE'S, SAFETY SSC'S				
- METHOD REVIEW	COMPLETE	2/86	3/86	1/86
- REVIEW OF LBEs/SSCs		2/86	4/86	N/A
MAJOR ISSUES				
DECAY HEAT REMOVAL	2/86	TO BE		N/A
CORE SUPPORT STRUCTURE	3/86	COVERED		N/A
REACTIVITY CONTROL	3/86	IN PSID,		N/A
FUEL	4/86	PRA, &		N/A
ISI	4/86	TECHNOLOGY		N/A
MULTIPLE MODULE CONTROL	5/86	PLAN		N/A
WATER/AIR INGRESS	7/85	"		N/A
STANDARD PLANT ISSUES	8/86	"		N/A
CONTAINMENT/CONFINEMENT	8/86	"		9/86
BOP CLASSIFICATION	8/86	"		9/86

ITEM	BRIEFING TO NRC STAFF	SUBMITTAL TO NRC	NRC ACTION	ACRS BRIEFING
DESIGN & TECHNOLOGY REVIEW				
BRIEFING ON REFERENCE CONCEPT	COMPLETE	-	1/86	1/86
TECHNOLOGY PLAN	4/86	9/86	6/87	N/A
PRA	N/A	9/86	6/87	TBD
PSID				
- OUTLINE	COMPLETE	COMPLETE	COMPLETE	N/A
- PSID SUBMITTAL	N/A	9/86	6/87	TBD
- LICENSABILITY STATEMENT	N/A	N/A	9/87	TBD

REVIEW SUPPORT

NRR TECHNICAL ASSISTANCE

MIT - FUEL DESIGN AND PERFORMANCE

ORNL - PERFORM INDEPENDENT ANALYSIS OF SPECIFIED ACCIDENTS AND TRANSIENTS. AID SPEB IN ITS REVIEWS AND ASSESSMENT.

BNL - SIMILAR TO ORNL, DEVELOPING A WATER/AIR INGRESS ACCIDENT ACCIDENT ANALYTICAL CAPABILITY.

RES SUPPORT

- COMPLETION OF HTGR HANDBOOK:
  - BROAD SCOPE-IDENTIFICATION AND ASSESSMENT OF AVAILABLE HTGR INFORMATION AND TECHNOLOGY
  - AIDS FSV REVIEWS AND INCIDENT RESPONSE
  - TRAINING TOOL FOR NEW HTGR WORKERS

FOREIGN TECHNOLOGY

- WORKING COMMUNICATIONS WITH ORNL
- THTR STARTUP AND HTR-500 DESIGN
- INTERNATIONAL CONFERENCES

PLANT DESIGN OVERVIEW
MODULAR HTGR

PRESENTED TO THE ACRS

JANUARY 30, 1986

W.R. SHERIDAN – SENIOR PROJECT MANAGER

STONE & WEBSTER ENGINEERING CORPORATION

DESIGN OVERVIEW OUTLINE

- KEY DESIGN SELECTIONS
- PLOT PLAN
- NUCLEAR ISLAND
- ENERGY CONVERSION AREA

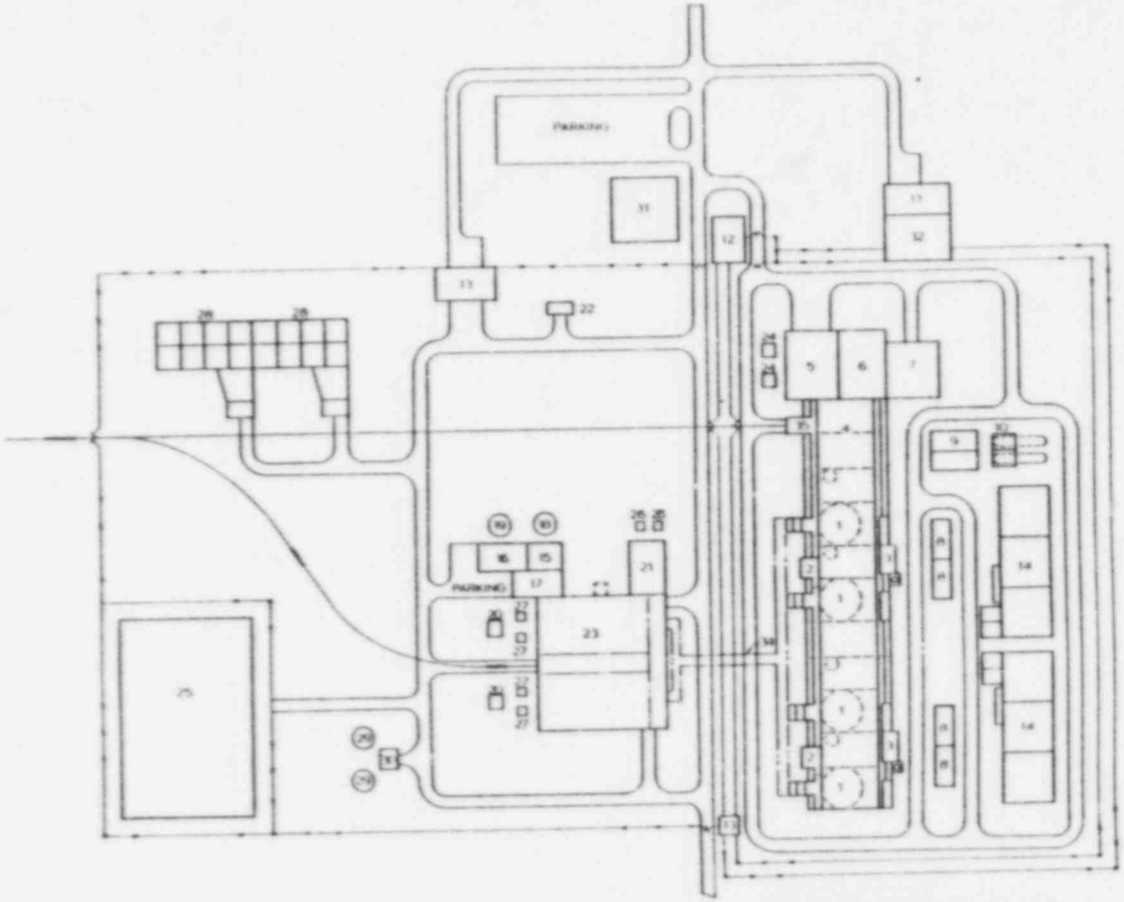
OVERALL PLANT
KEY DESIGN SELECTIONS

FOUR (4) x 350 MWt MODULAR HTGRs

TWO (2) x 300 MWe TURBINE GENERATORS

SINGLE CONTROL ROOM, MULTI MODULE CONTROL

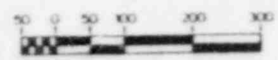
SEPARATION OF NUCLEAR ISLAND AND ENERGY
CONVERSION AREA



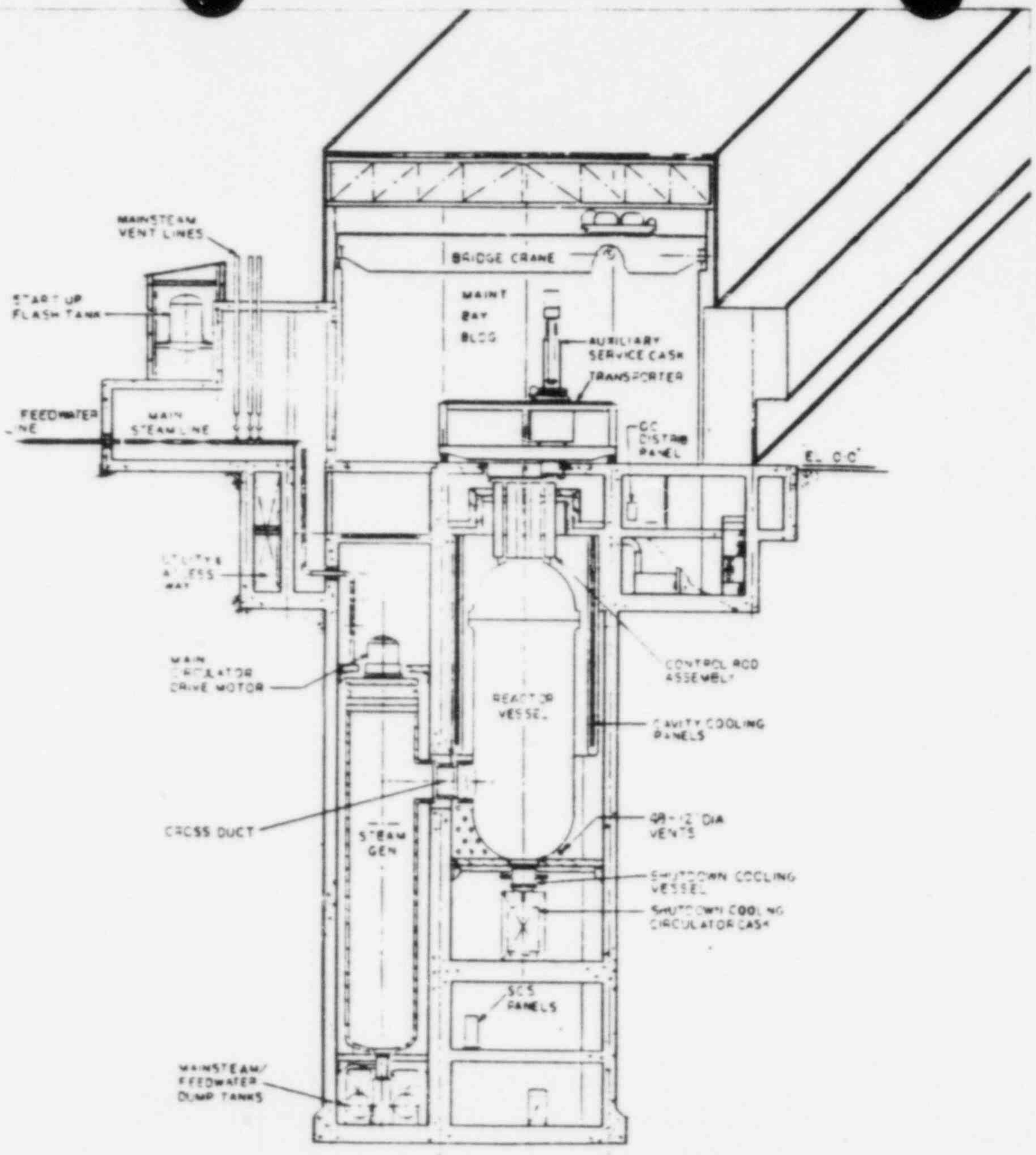
LEGEND

1. REACTOR BUILDING
2. REACTOR AUXILIARY BUILDING (WEST)
3. REACTOR AUXILIARY BUILDING (EAST)
4. REACTOR SERVICE BUILDING
5. CONTROL BUILDING
6. PERSONNEL SERVICES BUILDING
7. RADIOACTIVE WASTE MANAGEMENT BUILDING
8. ULTIMATE HEAT SINK STRUCTURE
9. STANDBY POWER BUILDING
10. FUEL OIL STORAGE TANKS
11. CARGO SEARCH AREA
12. SECURITY BUILDING
13. GUARD HOUSE
14. HELIUM & NITROGEN STORAGE AREA
15. AUXILIARY BOILER BUILDING
16. MAKEUP WATER TREATMENT BUILDING
17. MAINTENANCE BUILDING
18. CONDENSATE WATER STORAGE
19. DEMINERALIZED WATER STORAGE TANK
20. UNIT TRANSFORMER
21. NON ESSENTIAL SWITCHGEAR BUILDING
22. HYDROGEN STORAGE AREA
23. TURBINE BUILDING
24. ESSENTIAL BUS TRANSFORMER
25. SWITCHYARD
26. STARTUP TRANSFORMER
27. UNIT AUXILIARY TRANSFORMER
28. STATION COOLING TOWER
29. FIRE WATER STORAGE TANK
30. FIRE PUMP HOUSE
31. ADMINISTRATION BUILDING
32. NI WAREHOUSE
33. B.O.P. WAREHOUSE
34. MAIN STEAM/FEEDWATER PIPING
35. WASHDOWN BUILDING

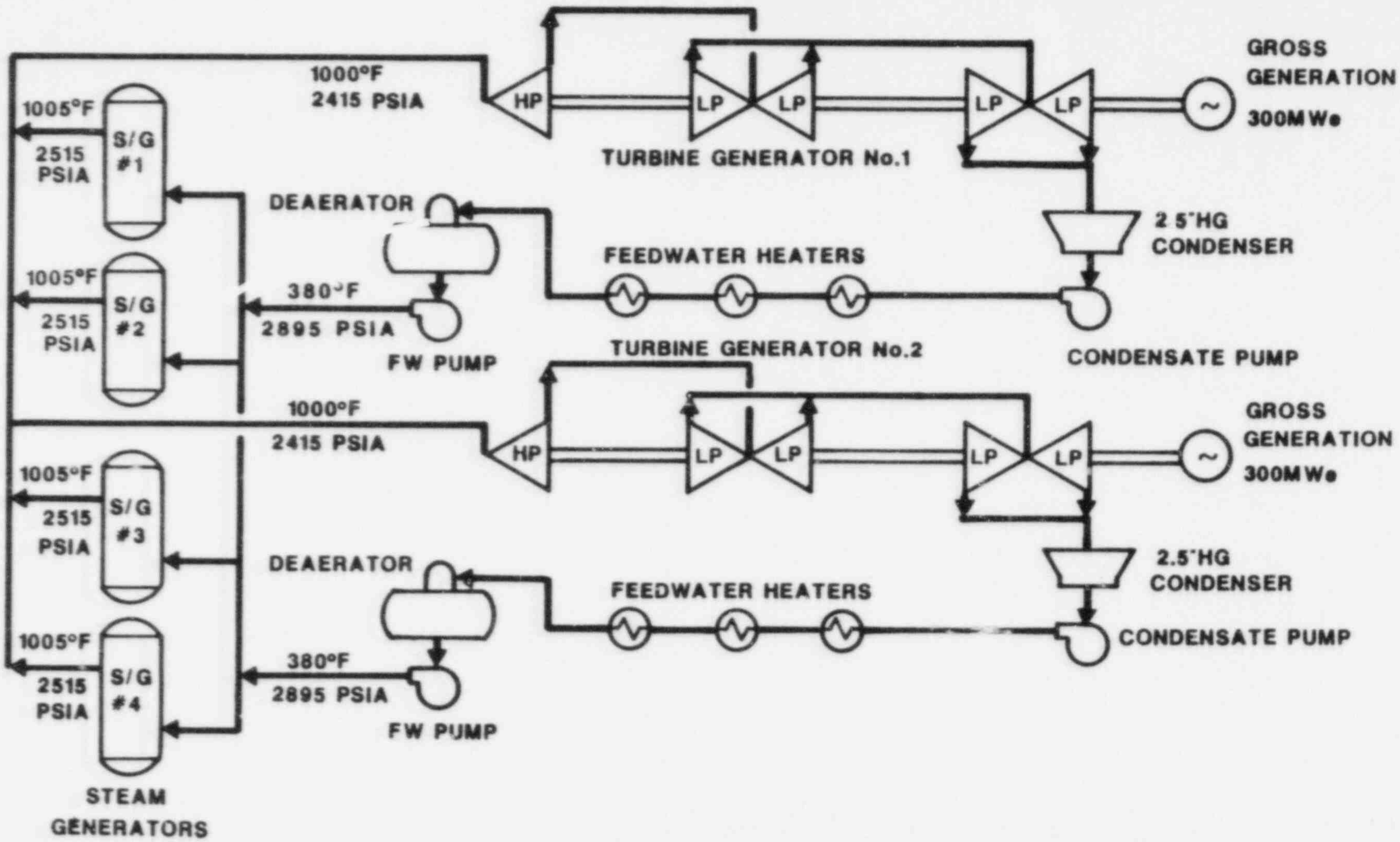
PLOT PLAN



REACTOR BUILDING ELEVATION



TURBINE CYCLE



PLANT CYCLE PARAMETERS

CORE POWER (4 x 350 Mwt)	1400 MW(t)
GROSS POWER TO TURBINE	1403 MW(t)
STEAM PRESSURE	2415 psia
STEAM TEMPERATURE	1000 F
TURBINE EXHAUST PRESSURE	2.5 INCHES Hg
TOTAL GENERATION	600 MWe
AUXILIARY POWER	42 MWe
NET GENERATION	558 MWe
NET EFFICIENCY	39.9%