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CALLAWAY UNIT 1 PLANT TOUR AND MEETING

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

CALLAWAY UNIT 1 PLANT TOUR AND MEETING

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End Zone Room
Holiday Inn West
1900 I-70 Drive, S.W.
Columbia, Missouri
Thursday, November 5, 1981

The meeting of the Advisory Committee on Reactor Safeguards as convened at 8:30 a.m.

PRESENT FOR THE ACRS:

M. CARBON, Chairman
C. MARK, Member
J. RAY, Member
J. ARNOLD, Consultant
W. LIPINSKI, Consultant
Z. ZUDANS, Consultant

DESIGNATED FEDERAL EMPLOYEE:
R. MAJOR

ALSO PRESENT:
For the NRC Staff and Industry:

G. EDISON
B. YOUNGBLOOD
D. SCHNELL
R. SCHUKAI
J. MC LAUGHLIN
D. CAPONE
A. PASSWATER
N. SLATEN
G. HUGHES
D. SHAFER
D. WALKER
R. DETTENMIER
M. STILLER
S. MILTENBERGER
A. NEUHALFEN
R. WILKES
P. APPLEBY
M. TAYLOR

ALSO PRESENT (continued):

D. HEINLEIN
W. WEBER
J. KAELIN
K. KUECHENMEISTER
R. COTHREN
F. FIELD
D. MC ALEENAN
J. WATSON
N. PETRICK
F. SCHWOERER
R. STRIGHT
E. BECKETT
J. CERMAK
J. SMITH
J. PREBULA
D. GASDA
K. LEE
P. WARD
D. GROVE
F. RODDY
E. THOMAS
R. FOX
D. RAWLINS
W. LUCE
J. SWOGGER

ALSO PRESENT (continued):

C. TULEY
D. PADDLEFORD
G. BUTTERWORTH
G. LAND
J. MESMERINGER
J. IRONS
S. SANCAKTER
J. GRESHAM
T. TIMMONS
D. BALLMANN
G. RATHBUN

P R O C E E D I N G S

(8:30 a.m.)

MR. CARBON: The meeting will come to order. This is a continuation of the ACRS Subcommittee meeting on Callaway Unit No. 1.

Today the subcommittee will continue its review of the Union Electric Company's application for an operating license for Callaway Unit No. 1.

Mr. Richard Major on my left is the designated federal employee for this meeting.

A transcript of the meeting is being kept and, as before, it is requested that each speaker identify himself or herself, and speak loudly.

We have not received either written statements or requests for time to make oral statements from any member of the public, and I would again ask: Is there anyone here who wishes time for comments?

(No response.)

MR. CARBON: I see no such indication.

We will now resume the meeting. I would like to suggest first to the subcommittee members and consultants that during the morning you be thinking about the topics that we would ask the applicant to address if we recommend that they come to the committee next week, both the topics and the time. If you would, jot these things down on notes and give them to me before the end of the meeting, we will try and put something together.

Does anyone have any comments to make, or any questions at this time?

(No response.)

MR. CARBON: Let's go ahead then. I will call on Mr. Edison to cover both the confirmatory issues and the licensing conditions, and then we will call on the applicant for a response to his comments in both categories.

(Slide.)

MR. EDISON: Thank you, Mr. Chairman. My name is Gordon Edison. I am the NRC Staff Licensing Project Manager, and my Branch Chief, Mr. Joe Youngblood, is the other member of the Staff here to represent the Staff.

(Slide.)

I am going to summarize the confirmatory items to the license condition that we found and placed in the Safety Evaluation Report for the Callaway operating license.

First of all, the summary will cover the confirmatory items from the safety benefit. What we mean by a "confirmatory item is that the Staff basically agrees with the Applicant's design, but that there may be a need to complete documentation or to verify some implementation at the site; perhaps the analyses are 95 percent complete and we want to make sure that the last 5 percent are what we think they are going to be, and the analyses will not change the result; or we want to look at some test results or records at the site.

(Slide.)

The status of the confirmatory items is that we found a total of 27 items, in addition to 12 TMI-related confirmatory items. Two of these items may be confirmed by the full committee meeting next week. Those items are Items Nos. 4 and 9 in the Safety Evaluation Report. Number 4 has to do with cladding collapse time; and number 9 has to do with operator actions for safe shutdown from outside the control room.

We have received submittals from the Applicant on both of those items in the last few days, and the Staff is now reviewing those submittals.

(Slide.)

I will summarize the license conditions that are in the Safety Evaluation Report, NUREG-0830.

(Slide.)

By "license condition" we mean some action that the Applicant must take, or a condition to be maintained over the operating license of the plant. An example of a license condition are the technical specifications.

(Slide.)

Section 1.9 of the Safety Evaluation Report shows some 20 license conditions. A number of these we expect to be implemented prior to licensing and fuel load. Therefore, these may cease to exist as licensing conditions once the Staff sees that they are in place and we are satisfied and they will no longer be a license condition.

I would like to say something about one of these license conditions. Number 17 refers to "separation barriers" between cabling. We indicated in the Safety Evaluation Report that we were going to require tests to substantiate the barrier. That is no longer the case. We do not feel that we need that test.

The Staff was discussing whether there would be a need for testing, or whether an on-site visit to inspect the separation and satisfy ourselves that it was adequate would be good enough, and we decided that the on-site visit was satisfactory. So there will be a change, and hopefully we can get that in the supplement that is to be published on November 27th to show that that testing is no longer required.

That completes the Staff's presentation on the confirmatory items and the license conditions. Unless there are any questions, I will step down and allow the Applicant to respond.

MR. CARBON: I wonder if you would take one of the confirmatory issues -- I would arbitrarily take number 7, for example, steam generator level control and detection -- and simply state what it is that the Applicant has not yet provided? I am asking this just for an example.

MR. EDISON: All right. Let me get my SER off the table there and look at what number 7 is.

(Pause.)

All right. This is an issue having to do with steam generator level control and detection. In its review, the Staff noted that three steam generator level channels were used in two out of three logic for isolation of feedwater high steam generator level. One of those same three level channels was also used for control.

This did not meet IEEE Standard 2.79, so the Staff asked that the Applicant modify the design. The Applicant did modify the design. They have changed to a two out of four logic. So we found that to be acceptable and in complicance with this IEEE Standard.

Now what we want the applicant to do is to modify the Final Safety Analysis Report's description of this, so that it is documented. What the Staff has done has been to-- I think this is one we have required as a license condition, and said the Applicant must modify the design, we must have this two out of four, and we are going to insist that that be there. And if you in fact implement that and build that into -- it is so important, that you build that into the design before the fuel load date. We will look at that, and we will take that off of the list of license conditions.

As a matter of a confirmatory item, however, we want to see that in the documentation for the rest of the public and the Staff in the future.

So we have a license condition that they must make the change to the hardware and the design.

Secondly, as a confirmatory item we want to see the documentation in the book.

MR. CARBON: So they have done it, and it is on this list simply because the paperwork has not been done?

MR. EDISON: That is correct.

MR. CARBON: Are there any questions?

(No response.)

MR. CARBON: Fine. Thank you.

MR. PASSWATER: Mr. Chairman, Bob Stright, the SNUPPS Licensing Manager, will respond to the confirmatory items.

(Slide.)

MR. STRIGHT: Good morning.

In general we agree with Dr. Edison's assessment of the confirmatory items and license conditions. We were going to note a couple of the confirmatory items that we have recently submitted information on, but Dr. Edison has already passed that information on.

Concerning Confirmatory Issue No. 9, that has to do with operator actions from outside the control room. I thought we might talk about this for a minute, and possibly take care of an agenda item on today's agenda for later on.

(Slide.)

The original issue came up because our FSAR had some words in there -- "jerryrigging" was the term that was used for extended hot shutdown from outside the control room. This brought up some discussion with the Staff about what actual steps were required.

We clarified that with the Staff, and removed the term "jerryrigging" because there are no special adjustments or unusual arrangements that have to be done. The FSAR has been changed.

Then we recently submitted a letter that detailed the steps, including boration, that would be required to maintain an extended hot shutdown from outside the control room. Concerning the agenda item on your agenda today I believe this issue came about because some other designs the Committee may have reviewed that had one train of instrumentation and controls outside of the control room at the remote shutdown panel. Therefore, with the single failure we could not guarantee that they could maintain this safe shutdown condition from the remote panel.

The SNUPPS design is different from that. The remote shutdown panel is actually two separate panels. We can sustain a single failure and still meet the functions that were intended for the remote shutdown panel.

That has been reviewed by the Staff and that is why it was not an open item on Callaway SER as has been on some recent plants.

MR. ZUDANS: Those two panels are in a single room. What do you plan to do then?

MR. STRIGHT: The two panels? They are in a single room right now. However, as part of our evaluation of the control room fire which we touched on yesterday, we are considering the possibility and the capability that the design exists for the fire barrier between the two remote shutdown panels.

That decision has not been made yet, but that's very likely a possibility.

MR. ZUDANS: I see. Thank you.

MR. STRIGHT: Concerning the license conditions, we agree with Dr. Edison that most of these items would not really appear as license conditions. A good many of them are duplicated from the confirmatory list and we believe these will be cleared up a long time prior to licensing.

I would like to make a couple of comments on two of the license conditions. Item number three is really an administrative matter. However, this matter states that there will be a license condition that we will implement, the secondary chemistry program that we described in the FSAR. We fully intend to do that.

We also fully intend to implement all the other programs that are described in the FSAR. We don't really understand why this one is called out special. But it is an administrative problem, not a real problem.

Concerning license condition number one, the Staff has a stated requirement to perform surveillance on the control rods. We don't believe at this time that the surveillance program suggested by the Staff is necessary. We intend to propose an alternative program and we will discuss this issue with the Staff further.

Other than those clarifications we don't have any problems with the confirmatory items or the license conditions and do not see that any of those are going to be hard to resolve or to implement.

MR. CARBON: Questions?

MR. MARK: On the matter you mentioned on the control rods, what is it that the Staff has proposed which seems to you undue and is hapnium control rod a new thing? Or is it just another example of a sort of control that has generally been used before?

MR. STRIGHT: Your first question, the Staff suggested a visual surveillance at the beginning of the first, third, seventh, and ninth, or something like that, cycles.

Let me back off now to the second question. Hapnium used in control rods. There is a considerable wealth of experience using hampnium in control rods. However, most of that is not in the commercial nuclear plants. There is some experience with commercial nuclear plants.

However, recently Comanche Peak and both of the SNUPPS plants are the first ones to recently shift from silver indiancadmium to hapnium. The Staff required a survey surveillance program at the Comanche Peak and those applicants agreed to institute this program.

We don't feel it's necessary and especially if Comanche Peak does it we're going to be able to get information from them.

MR. CARBON: Fine, thank you.

MR. PASSWATER: Mr. Chairman, an administrative thing. On the agenda, as Mr. Stright mentioned, there's an item called Interpretation of DDST. We really don't have anything to add beyond what he said unless there are some questions that the Committee has.

MR. CARBON: Any questions on that?

(No response.)

MR. PASSWATER: So we won't have a separate presentation on that item.

The next item, then, is the preparation of emergency operating procedures. Andy Neuhalfen, who is the Superintenient of Operations, Callaway, will talk on that topic.

MR. NEUHALFEN: Mr. Chairman, my name is Andrew Neuhalfen. I'm the Superintendent of Operations, Callaway Nuclear Plant. Part of my responsibility is for the implementation and development of the emergency procedures for Callaway.

(Slide.)

A brief outline of what I had planned to cover this morning is the current development of our emergency procedures. I would like to delve into the philosophy of the format. I would like to show you an example of just a page from one of the procedures. We went to the procedures structure. Then I plan to show you lists of the guidelines themselves.

We have discussed the status trees in yesterday's presentation. I have a little bit more to add to the status trees.

Then we go into the use of the coordinated procedures along with the status trees. I am prepared to show you an example of the dianostic procedure and a brief summary.

(Slide.)

We are currently using the guidelines developed by the Westinghouse Owners Groups. These are written generically for Westinghouse four-loop, twelve-foot core plants, 25-35 watts. This provide a comprehensive and integrated set of emergency guidelines along with related background information for each procedure.

In addition, there are analytical bases and training and application information, again for each procedure. Included is also a preparation and users guide for writing procedures.

These guidelines address the immediate procedures for addressing transients and emergencies and long-term requirements. This gives us the assurance of operator preparedness for events up to and beyond the design basis event.

The development of emergency procedures gives consideration to normal patterns of training operators when confronted with a plant upset condition. These include, of course, the immediate actions, immediate verifications, and the system checks. Consideration is also given to the usefulness of a procedure to an experienced operator versus a relatively inexperienced operator, given the fact that the procedures are not overwhelming to an experienced operator.

They serve as a guide to the operator when verifications of manual or automatic actions cannot be obtained. They provide for smooth transitions between the guidelines and the contingencies They minimize the impact of adding new contingencies to an existing procedure. They also allow us to reference the alarm responses into specific contingency procedures.

What this will do for us, then, is present a plant-specific procedure which adheres to the human factors concept in facilitating clear understanding and transfer of information to the operator under stressful conditions.

An example, then, of the format we are using. It is a two-column format. The lefthand portion of the column is the action and expected response. The righthand portion is to be used if the action or expected response is not obtained. It is in the righthand portion of the procedure, but in a column where we refer the operator to his contingency actions. We spoke of this yesterday.

As you can see, each page is blocked out. This will be on the righthand side of the page to allow for binding on the lefthand side. Each page has its own title, guideline title, series, current revision date and the procedures utilized, very strategically placed words, cautions, as you can see, and notes precede the step rather than being steps to be performed which are advisory.

Format E is using strategically placed words -- if/then, less than/greater than -- in the format and they are very consistent throughout. All of our abbreviations are standard abbreviations also.

As you can see, we place only steps within the guidelines. There is little, if any, background information. This is all provided to our operators in their training program. Of course, it is available for each procedure, but it is not in the procedure itself.

MR. ZUDANS: Could you qualify the distinction between guideline and procedure?

MR. NEUHALFEN: Yes, sir. These are the guidelines that we will be using to write our procedures. They are Westinghouse Owners Group guidelines.

MR. ZUDANS: I see.

(Slide.)

MR. NEUHALFEN: The procedure structure is divided into two sets. The first set consists of two parts: one, emergency procedure for diagnosis and mitigation in the classical sense; number two, emergency subprocedures with additional steps applicable to a particular emergency procedure; number three, emergency contingency actions applicable to more than one emergency procedure.

The second set utilizes successive halves, as we saw yesterday, to provide a systematic approach and hierarchy of protection for the operator to use to mitigate consequences of any accident that jeopardizes the nuclear safety of the plant.

The second set consists of two parts: the critical safety function monitoring, using status trees; number two, the restoration guidelines to address loss of the critical safety functions.

(Slide.)

The optimal recovery guidelines are contained in this list. As I say, the emergency procedures are designated in the E series. They cover loss of reactor coolant, of course, the diagnostic EO and steam generator tube rupture. There are, in addition, emergency subprocedures which can be referenced or are referenced from the E procedures.

(Slide.)

The emergency contingency actions, of which I spoke previously, are the anticipated transients without scram, loss of AC power. This emergency contingency action has a recovery with or without safety injection required. Also, the steam generator tube rupture contingencies, which cover multiple tube failures within a steam generator or tube ruptures within more than one steam generator.

MR. LIPINSKI: On the loss of AC power recovery with safety injection, how do you accomplish safety injection without AC power?

MR. NEUHALFEN: In the background information this is accomplished through the use of bleed and feed, if I am not mistaken. Mr. Lang is here to delve into the actual background information on that procedure.

MR. LIPINSKI: That will come in later in our agenda? I'll delay questions, then, until then.

MR. NEUHALFEN: Thank you.

(Slide.)

The function restoration guideline set will be available in mid-'82 through the Westinghouse Owners Groups. These are used in conjunction with but are not limited to use with the status tree. These are, as we saw yesterday, referenced from the status trees out into another procedure. They take care of the subcriticality critical function, the reactor coolant system integrity, reactor coolant -- no, this is inadequate core cooling -- reactor coolant inventory, loss of secondary heat sink and also the

Z series, which is the containment critical safety function.

(Slide.)

An example, we saw the one on inadequate core cooling yesterday, this is an example of subcriticality. as Mr. Cermak pointed out to us yesterday, they are color-coded. Plants are using them in the SPDS, red meaning a critical safety function has been violated and you have to take immediate action.

Orange is the critical safety is in jeopardy, subsequent action is necessary. Yellow means the critical safety function has not been satisfied; actions will be necessary on the part of the operator. Green, of course, means that in this case subcriticality critical safety function has been satisfied.

We enter the status tree from the left. I hope the writing is large enough so that you can read it.

MR. ZUDANS: I have a question. From what I gather up to this point you have two sets of procedures. One is that it directs you to the set you call optimum procedure or optimal recovery. The other one is based on the status trees, which you call functional recovery.

Is it to be understood that the first step would always refer to the case where you know for sure what's happened?

MR. NEUHALFEN: I think in my next slide I can address that for you, if I might, sir.

MR. ZUDANS: Fine.

(Slide.)

MR. NEUHALFEN: The next one is the coordinated use of the procedures. It shows, of course, when an abnormal condition is encountered, response is required by the operator. The response he takes, he takes immediate action. He restores the condition. Of course, he goes back to normal operation.

Should it not, we then encounter a dual path where we are monitoring the critical safety function trees as well as going into our emergency procedures. In this way we are affecting recovery through the optimal recovery guidelines or procedures in order to get back to normal operation.

Should, during the course of carrying out the emergency procedure, one of our critical safety functions becomes in jeopardy, then by monitoring our SPDS we can refer into one of the function restoration guidelines and get the plant back into a safe mode as far as critical safety functions are concerned and recovery the plant using the optimal recovery guidelines.

Does this answer your question?

MR. ZUDANS: Yes, it does. In this critical safety function you have identified six items there?

MR. NEUHALFEN: Yes, sir.

MR. ZUDANS: For which you had these critical function status trees.

MR. NEUHALFEN: Yes, sir.

MR. ZUDANS: It seems to be clear but not quite completely when you really branch. Do you branch in every case where you cannot restore the condition that you show on your chart? Do you branch in both directions?

MR. NEUHALFEN: Yes, sir.

MR. ZUDANS: And this monitor on SPDS would just flash different in one of your six states and scan through them and see whether any of these six functions is affected, right? In the meantime, the emergency procedure would be followed on the basis of some other deterministic indication?

MR. NEUHALFEN: You are absolutely correct, sir. But what I am trying to point out here is that we are not going into the procedures blind. We are always continuously monitoring our critical safety functions in order to cover any contingency that might not be covered in our emergency procedures.

It also allows us to maintain an ever-watchful eye on our plant.

MR. ZUDANS: How do you decide? Say, look at your emergency procedures block. How do you decide to get on this "no" path that you indicated?

MR. NEUHALFEN: This one here?

MR. ZUDANS: Right.

MR. NEUHALFEN: If we are not -- okay, if the emergency procedure is not covering --

MR. ZUDANS: How do you know that?

MR. NEUHALFEN: We can tell by the indication in the control room. The trained operator, of course, is going to be looking at his control board and his parameters that he is trying to recover. If he's not able to recover the parameters using the emergency procedures, his critical safety function trees and the SPDS will be telling him where he is in core cooling, et cetera, any one of a number of six.

So he then refers to the function restoration guidelines in order to recover the plant to a safe condition so that he can effect repairs and go back to normal operation again.

MR. ZUDANS: Maybe it's clear to you, but you're not really conveying it competely clearly to me. Maybe it's not important. I think what you have is a good thing, but there are a few links missing, at least to convey to me. Maybe others --

MR. LANG: Dr. Zudans, the idea is if you get into one of these E-1, 2, 3, guidelines, which is an event-specific procedure, if you are in that procedure and you're not getting the expected response the operator would be trained to see then, he would kick out of the event-specific procedure and go back to his critical safety functions until he can rediagnose or find out if an additional failure occurred.

MR. ZUDANS: That's exactly how I thought you would answer my question. The point is, when you follow emergency procedures you know what you are trying to do. When you find out that that is not leading you to the expected goal, then you have this other resource to go and start looking at your event or status event tree.

MR. NEUHALFEN: That's correct.

MR. ZUDANS: So that's when you go on that "no" path. But the way you show in that procedure, the monitor SPDS is the only one that has the status trees. I thought you had duplicate pictures you could do at the same control room location also following the status trees. Do you have to go to SPDS, wherever that is, to get that done?

MR. NEUHALFEN: We also have the instrumentation that is on the control board.

MR. ZUDANS: Physically where is this monitor SPDS located?

MR. NEUHALFEN: At this time we are still evaluating that.

MR. PASSWATER: It's in the control room. The SPDS will be --

MR. ZUDANS: It's recallable on any of your CRTs which are in the control room or tech support center?

MR. PASSWATER: The final design of whether we're going to use the CRTs that you saw in the control room yesterday or whether we're going to add CRTs, one or two, is still being decided. There is some consideration about matching the software and so forth in this system and the existing system, but it would be available in the control room to the operator. It is not that he has to go somewhere else to get information from the SPDS.

MR. ZUDANS: The only thing is, the way this graph shows here, the only unit that would have that capability is the SPDS. I thought the SPDS was in addition to whatever else you have in the control room.

MR. PASSWATER: As I said, that's still being decided as to whether or not the capability to call that up will be on the existing CRTs in the control room or whether we're going to add another panel in the control room.

MR. ZUDANS: I would just like to add this is the first sensible thing I see in trying to do a nice job, so maybe you should make clear that each of the little items is fully understood.

MR. LIPINSKI: Let me clarify something for myself. The SPDS, the safety parameter display system, that NRC is asking for is simply a display of the critical values in the plant. It has nothing to do with your trees. The trees are not being asked for by the NRC in the SPDS.

MR. ZUDANS: They are proposing something else.

MR. LIPINSKI: The way this diagram shows it, the way the operator consults the SPDS and finds out what the status of the plant is, then he himself is going to make decisions on whether these parameters are proceeding in the right direction or not.

MR. ZUDANS: We agree on that. That's why I said I would like to see the operator do it in another box.

MR. ARNOLD: I am getting lost. Could you just run through a simple scenario which involves a person-to-person interaction? You've used the word "we" and "us" and "you". Who's doing all of these things? Who is making the responses? Is it just the operator that's responding or is someone else responding to the operator? What is happening?

MR. NEUHALFEN: In our control room we have a supervisor who is on shift. We have currently two reactor operators stationed in the control room along with him. In an emergency situation there is, of course, continuous communication between the operators and the operating supervisor.

When a situation comes about that we need to get into the emergency procedures, the procedures are followed both by the operator and the operating supervisor. The supervisor, of course, is fully cognizant of what is going on in the big picture in the control room, so he is the one who is picking up on something that is an unexpected response and then leading the operator on the right path.

Does this answer your question, sir?

MR. ARNOLD: I guess so, if you tell me that the response really is the responsibility of the supervisor. He's in charge and he is making all of the decisions.

MR. NEUHALFEN: He is not. He is making the decisions. However, he has licensed people giving him the input he needs to make the decisions.

MR. ARNOLD: I appreciate that. I'm just trying to understand who is involved here. It's just the operator.

MR. NEUHALFEN: Yes, sir, the way we envision it.

Were there any more questions on this slide?

MR. ZUDANS: The only final remark I would like to make, it's not a question exactly, not a concern, my objection, if one would want to label it that way is here you have other people to assist the operator. I would like to see the same capabilities you say in SPDS also exists in the operators. It's at the operator's direct disposal. He uses this only as a standby, in an assisting capacity rather than a primary capacity.

The way you show this chart he goes through normal optimal recovery procedures by himself. Then he hits this need to address the status trees. He has to go and call up the SDPS, which in my opinion should not be the case. He should be able to do the very same thing the SPDS can do but the SPDS only acts as an assisting unit and not as a principal unit of control.

That is just a thought now.

MR. NEUHALFEN: Along with the electronic display coming about through the SPDS we have a backup method. I had not planned to get into them, but we have backup methods of monitoring the critical safety function status trees. SPDS was a convenient term to put in this block because we do plan to use it and I see where your concern lies.

As I said, our operators are trained to look to these critical safety functions themselves. As I said, this is a backup for them. This is a backup reminder.

MR. LIPINSKI: One further question. The block contingency action, does this imply written material or are these decisions to be performed by the operator based on the conditions he observes?

MR. NEUHALFEN: These are the function restoration guides that are keyed from the safety parameter display system.

MR. LIPINSKI: Okay.

MR. ZUDANS: C-1, C-2, C-3.

MR. LIPINSKI: You ought to label the blocks so that there is a one-to-one correspondence between the other material.

MR. NEUHALFEN: Excuse me?

MR. LIPINSKI: I said you ought to label that block so that there is a one-to-one correspondence with the previous presentation.

MR. NEUHALFEN: My next slide are examples.

(Slide.)

What I want to do is briefly go down the precedure and show how we use it. We, of course, are verifying reactor trip. The procedure, very briefly, tells how to do this. This is, of course, the action or expected response. Should we not obtain this response, then we manually trip the reactor and go to the emergency contingency action, which is ATWS, verify turbine trip, which is step two. If the action response is not routine, then we manually trip the reactor.

Of course we can follow on down AC, initial safety injection has activated, verify feedwater, verify all the way down through on the contingency side, the righthand column.

Then we go to the emergency contingency actions or emergency subprocedures.

(Slide.)

I have one more slide. The circled items are immediate operator actions, circled for training purposes right now. As you can see, we are still lacking the background information in the procedure, which gets rid of its bulk. It is strictly the steps that the operator must go through in order to restore his plant.

MR. ARNOLD: A question. Have these instructions been tested on a simulator or validated in some other way to be sure that indeed when you get an indication of a problem that you are including in your instruction all of the possibilities or all of the sources of such an indication of a problem?

MR. NEUHALFEN: Sir, we are in the development stage now. It is our intent, when we get our simulator on-site, to write the procedures. We will then run them on our simulator, debug them on our simulator, and at that time we will have our procedures completely developed. But we do intend, I hope this answers your question, to debug them on our simulator.

MR. ARNOLD: And does your simulator software -- well, it just covers certain accidents, does it not, and certain responses and so on? What can you do or what have you done to further validate that this is a complete analysis of all of the probable sources of the indications of trouble that you are listing?

MR. NEUHALFEN: I will say that our simulator is current state-of-the-art. It has some, is it 800, some phenomenal amount of --

MR. ARNOLD: I'm talking about the input to the simulator rather than the simulator itself.

MR. NEUHALFEN: You are talking about the software for it.

MR. ARNOLD: The information, the knowledge you are putting into it. How do you know that you have all the knowledge necessary to give a set of instructions that is as definitive as these and know that you are not missing some important --

MR. LANG: We fully recognize that the capability of the simulators today do not fully check out with the precedures. However, what we want to verify using the simultators is the interaction between the individual control room monitoring the critical safety functions and those that are actually looking at the plant-specific optimal recovery guidelines.

We think that we can verify the interaction between these two individuals, that we have gone a long way toward verifying this concept.

MR. ARNOLD: I recognize that it's a very probable tool, but at the same time I recognize that there is nothing so disruptive as wrong information or wrong interpretation of a signal and a surprise to the operators as to the response to the corrective action.

What I am trying to understand is how completely you figured you have analyzed all of these situations so that when the fellow looks at the chart he makes a move to drop the temperature, it rises. That's just devastating. I don't know whether I'm making myself clear or not.

MR. LANG: Yes, sir. Again, I guess I would reiterate that we recognize that we cannot begin to pretend that we've considered every scenario that could occur in a power plant or an operation. However, we think that with these critical safety trees that if a situation does occur that takes you beyond an expected response where you cannot diagnose an event-specific event, we fell very strongly that the critical safety functions and the function recovery guidelines will lead us on a safe path.

As far as the instrumention and getting an erroneous signal, the implementation of Reg Guide 197 addresses ambiguity, for instance. It will give you an indication that one parameter is being given erroneous to an operator. If you have two indications, the Reg Guide will require a third to determine which one is correct.

MR. ARNOLD: That is to the extent of the power of the simulator as it is today. I guess what I'm trying to see is, do you have an ongoing program that will continue to expand the knowledge that is in the simulator?

MR. RAWLINS: Yes, sir, we do. We have just completed construction and it's currently in operation in a new facility in Pittsburgh, in which we have a simulator tied in. The development and research effort is continuing to put such things such as two-phase flow-in.

Going back to your original question, I would also like to bring out the point of view that in the development of these guidelines that it is not just a Westinghouse alone effort. It is a cooperative program through the Owners Group in which we have thirty, forty utilities involved with their operations people. With that collective experience, not only from U.S. plants but our international plants also, I think it goes a long way toward testing these out with people who have actually operated the nuclear power plants.

We also try to bring in not just the operations but the interdisciplinary aspects where we have accident analysis people, training people, human factors experts working on that subcommittee to develop these.

MR. ARNOLD: You are saying you do have feedback from the performances of simulators throughout the world?

MR. RAWLINS: Yes, sir.

MR. ARNOLD: Tell me, how does that feedback get back to the operators who are not involved in a particular incident?

MR. RAWLINS: It's fed back -- first of all, the procedures comes in to the Westhinghouse Owners Group. Periodically there are training seminars. I believe -- Glenn, you can correct me if I'm wrong -- the last one was a week-long seminar held in September at which the procedures and the bases, all the information behind what you see here, is explained to the operations staff of the various utilities. They are brought in and actually given seminars on the steps and why those steps are there.

MR. ARNOLD: Do you have any emergency rules in case you get something that is really vital knowledge for all of the other simulator operators? Do you pass that on quickly without going through seminars?

MR. RAWLINS: Yes, sir. If we find any validation of a bases for which the plant is licensed, we have our Safety Review Committee and its reporting structure to have the information out to any of the operators, plus the fact that within Westinghouse we have also created an emergency response facility of our own which any of the utilities has direct immediate access to, and we can assemble within our command structure and technical command center the necessary people to deal with the problems that come up that maybe we have not covered here.

MR. ARNOLD: Fine. Thank you.

MR. LIPINSKI: Could you back this slide up to the previous one?

MR. NEUHALFEN: Sure.

(Slide.)

MR. LIPINSKI: At the very bottom you've got verify auxiliary feedwater pumps running and then on the next one it shows verify the valve alignment. The basic function of interest is auxiliary feedwater flow. This is where trouble developed on TMI. The flow itself was not directly verified.

You are giving me two steps to try to verify that the flow is there, but if I've got a blockage in the line, then I've got an anomaly. I'll have my pump spray and I'll have my valves open. I still may not have flow, if I don't have a direct measure of flow displayed to the operator. He could verify both of these and still not have the flow condition satisfied and stand there and scratch his head.

MR. NEUHALFEN: If I understand your question correctly, sir, you are wondering -- well, let me go through how it works.

You are verifying that the pumps are running, first of all, both the motor-driven and the turbine-driven.

MR. LIPINSKI: Right.

MR. NEUHALFEN: Then on the next one we're verifying alignment. Part of our normal operating procedure is to be in the ESF lineup at all times. We are here verifying alignment and you're saying somewhere later on we've verifying flow or not?

MR. LIPINSKI: I am saying the flow is a function of interest. This procedure does not identify flow as the item of interest. You then get into subtasks that should identify flow, but if I have an anomaly of a blockage, then he doesn't look for the flow directly. He satisfies your requirements, but he still does not know whether he has flow or not.

MR. PASSWATER: Andy, if I could interject something here. This is a generic Westinghouse guideline or procedure. On Callaway we do have instrumentation to verify flow from auxiliary feedwater flow instrumentation.

MR. LIPINSKI: What bothers me is to see this as a generic output. After all, it happened at TMI. There was a lesson to be learned, and that is that the flow was the item of interest. The flow was up. The operators observed that. They were happy, but they didn't look to see whether they had flow.

MR. LANG: If you look down the procedure, sir, you will see on step 15 that you do verify aux feed flow. The reason for this, you must be setting up zero, which is the initial diagnostic issue. You must go through the immediate verifications of your safety function. A very quick pass over the control board verifies that all your safety equipment came on.

After the operator finishes that phase of verification he then goes and looks at the specific flow of the safety systems and you get to that in step 15. It says verify aux feed flow. So it is a logical way that we were told by the feedback of the Owners Group that an operator should function in the control room and that was later put in.

MR. LIPINSKI: No further comment.

MR. NEUHALFEN: Gentlemen, in summary, we are writing our emergency procedures to the Westinghouse Owners Group guidelines. I have attended the seminar in which Mr. Lang was relaying his information the last weekend in September, the last week in September. We have participated along with thirty other utilities in the development of these guidelines.

Our procedures will be plant-specific. They will adhere to human factors engineering. They will be useful. They will be of extreme use to an operator even under stressful conditions through the structure of the guideline. Our operators will receive a thorough understanding of the procedures through their training and simulator use.

MR. ZUDANS: On these procedures, in particular the functional recovery procedures, I don't remember whether you showed the slide or not when you listed those six particular aspects.

MR. NEUHALFEN: The critical safety functions? I did not, sir. That was function restorations.

MR. ZUDANS: The next slide, not this one.

MR. NEUHALFEN: I do have another slide.

(Slide.)

MR. ZUDANS: These are the six particular states that you have as status trees. You either have or will have status trees worked out?

MR. NEUHALFEN: Yes, sir.

MR. ZUDANS: These are the ones you will call on on your CRTs in a flow chart form in the color just like you showed?

MR. NEUHALFEN: Yes, sir.

MR. ZUDANS: The question is, how much of the rest of the procedures will be in computer storage so that they will flash up automatically and tell the operator that here you have these choices from what the parameters are that would indicate that you should either follow this optimal recovery procedure or you should stick to the functional recovery? It will guide him rather than looking for the looseleaf or whatever he might keep those papers in.

Is that someplace in the future in your mind?

MR. NEUHALFEN: Sir, it could be done. I am still under the current philosophy that we will have hard copies of our procedures. We are expanding our computer for the plant. This is a possibility. I have thought about it, but we still plan to go with hard copy at this time.

MR. CERMAK: I would like to amend what Andy said in that the procedures are not going to be on the CRT. They will be hard copy. However, we have also developed, in addition to the status trees and in addition to all the procedures, an accident identification diagnostic system which will lead you to which accident you are involved in.

We haven't discussed that in detail. It's still under development with a group of twelve BWR facilities, of which SNUPPS is a member.

MR. ZUDANS: This accident diagnostic identification system, that, of course, is a computerized system.

MR. CERMAK: Yes.

MR. ZUDANS: Why don't you talk about it?

MR. CERMAK: Because it's not fully developed at this point in time. We are in the process of developing it. It's going to be tested on the Indian Point simulator next January and we will be making full submittals to the

NRC at that time.

MR. ZUDANS: That's exactly what the industry needs.

MR. CERMAK: That's what we're looking on.

MR. ZUDANS: Nice to see that.

MR. NEUHALFEN: Further questions, gentlemen?

MR. LIPINSKI: No, but I would like to make a comment to the Chairman. This relates to the Callaway review now, but generically I think this information is of direct interest to the ACRS in terms of how industry procedures are proceeding. I'm not sure where it fits within the Committee structure, but I think the Committee should take a close look at what is being done.

MR. NEUHALFEN: Gentlemen, I have a full set of procedures and guidelines as they have been written with me, and you may, of course, look at them if you care to, as well as all of the critical safety function status trees. I have all of the information with me.

MR. CARBON: Thank you. Any other questions?

(No response.)

MR. PASSWATER: Mr. Chairman, just in the interest of not leaving any misunderstanding, there was a question by Dr. Lipinski earlier about the safety injection being required in loss of all AC. I thought there might be a misunderstanding that there was going to be another presentation.

MR. LIPINSKI: They made reference to feed and bleed and we're going to talk about feed and bleed last on the agenda, I believe.

MR. LANG: What that entails is recovery of the event with and without. If you restore AC, how do you recover with and without? That is the intent.

MR. LIPINSKI: I don't get that from the single line statement.

MR. MILTENBERGER: Steve Miltenberger, Union Electric. In the outline it had three items listed. It had ECA-2. That was loss of all AC. That's the overriding procedure called for loss of AC and what to do about it.

The next item was ECA-2.1, subtier of a main procedure. 2.2, as was mentioned, the 2.1 and the 2.2 are subtiers of a main procedure that go into the recovery. So the main loss of all AC is covered in the main procedure, and then when you get into where you get AC back you get proper guidance to restore power directly.

MR. PASSWATER: Our next topic is the design of control room. Mike Taylor, Assistant Superintendent of Operations for Callaway, is going to speak on that topic.

MR. TAYLOR: My name is Mike Taylor. I'm the Assistant Superintendent of Operations for the Callaway plant and I'm going to discuss the control room design used at Callaway.

Initially in the SNUPPS concept Bechtel was tasked with designing the control room for the Callaway plant as well as all the SNUPPS plants. The concept to be used was a centralized control room with a pitch board, vertical board, enunciator panel on the read, a portion of the control panel, as we saw yesterday in the control room, and also a lower section benchboard and incline board on the front section.

The controls and indications were divided among these different boards with the front portions utilizing controls that would be most frequently used by the operators during normal and abnormal situations.

Bechtel produced this design and submitted it to the various utilities. As we heard yesterday, two of these utilities, Rochester Gas and Electric and Northern States Power Company, had operating nuclear plants at that time and could provide valuable input to the evaluation of this design. In addition, Union Electric Company had had over twenty years of design work in centralized control rooms.

The design was evaluated by plant review groups in these utilities and upon finalization of the design a wooden mockup was constructed. This mockup was a full-scale mockup so that operators and various other personnel from the utilities could evaluate this result.

Some of the things that were evaluated by lic ..sed operators from both the Prairie Island plant and the Ginna station were the following: the man/machine interface; video displays, two on the rear panel, two on the front panel; the actual control layout; functional grouping of controls; switch placement sizes; labeling; enunciator systems; and actual access to the boards themselves.

One of the features that came about as a result of this review was very noticeable. You may have noticed yesterday the gap in the front panels which allows easier access to the rear board from the operation stations. Bechtel then incorporated these changes that were recommended into a final design and the Callaway simulator was constructed at Zion, Illinois, at the Westinghouse nuclear training center. This was the first SNUPPS simulator.

As was stated yesterday, this was a Callaway simulator. The one portion on the left rear panel is a site-specific portion of the board. That one at Zion is the same as Callaway as well the new simulator is the same at Callaway. Our simulator will be the third to be operated of this type. It's undergoing further testing at Pittsburgh right now.

All of our operational personnel that will be licensed have gone through certification training on the simulator at Zion, Illinois. They have provided additional input as far as the design and use of this control room and the design of the control board.

I would like to take a few minutes here to just go over some of the features on the control board.

MR. CARBON: When was the simulator at Zion built and started?

MR. TAYLOR: I was first designed in 1979. Our first plant went through in 1979 and I believe it was the second or third class to use that simulator.

MR. CARBON: When was the design on it frozen?

MR. TAYLOR: I'm not sure of that exact date, sir.

MR. CAPONE: I would say about a year prior to that. Of course, changes that could be fed in were fed in as we were going along, but at some point you have to freeze the design.

MR. RAY: Evidently, then, this review process and the evolutionary deliberations that took place were before TMI.

MR. TAYLOR: For the most part, yes, sir.

MR. RAY: Have you made any changes since TMI in what you are projecting to do?

MR. TAYLOR: There are some changes to the control board since TMI. Those will not be shown on this slide because these were taken at the simulator at Zion just for the ease of demonstration.

MR. RAY: But you did update the design to encompass the lessons learned from TMI?

MR. TAYLOR: Yes, sir.

MR. CARBON: I had forgotten from yesterday where the board is split. Would you point that out?

MR. TAYLOR: Right here, sir (indicating).

MR. CARBON: That's for access to the rear?

TAYLOR: Normally the two operators would be in this position and this position over here (indicating). Turbine generator steam volume control system, and for ease of access back to the rear boards right there.

(Slide.)

This is a view looking from the left side of the control room.

(Slide.)

And from the right side. You can see the spacing here between the front and rear panels. Also we can see how an operator stationed at the front panel can look over the back of this panel and observe indications of controls on the rear panel also, as well as the visual enunciators around the top.

An operator can acknowledge enunciators on the front panels as well as the rear panels.

MR. ZUDANS: I guess I must recall incorrectly. I thought you have four CRTs on the main panel.

MR. TAYLOR: Here and here.

MR. ZUDANS: On the front. Just two?

MR. TAYLOR: There are separate keyboards for the two pairs.

(Slide.)

MR. RAY: Question. I think you said there was consideration being given to an additional CRT. Where will that go in that previous diagram?

MR. TAYLOR: That has not been decided as yet.

MR. RAY: But it will be in this area presumably.

MR. TAYLOR: Some of the options being considered would be maybe an extension on one of the front panels.

MR. ZUDANS: So what you are showing is at Zion?

MR. TAYLOR: This is at Zion, but for our purposes this is the Callaway design.

MR. ARNOLD: Yesterday I believe we were told, in answer to a question, that all fourteen of the shift personnel had their major point of presence in the control room. Is that wrong?

MR. TAYLOR: The two --

MR. ARNOLD: If so, where were the other two?

MR. TAYLOR: Normally the control room would be manned by two licensed reactor operators and one licensed senior reactor operator. That is the normal manning. There would be another licensed senior reactor operator on site. He, too, will be in the control room.

MR. ARNOLD: I guess I misunderstood yesterday. Could someone clarify that?

MR. STILLER: Let me clear that up. The comment was made, where are the fourteen people?

MR. ARNOLD: Exactly.

MR. STILLER: The fourteen people are all dispatched from this control room. At some time during the shift they may or may not be at the control room. We have two there at all times. Where they go --

MR. ARNOLD: Where do they gather in the control room? I'm just trying to understand.

MR. STILLER: They don't gather. They don't assemble in the control room. They may be there. When a man shows up on watch he is relieving someone. What is that man doing that he is relieving? The water treating plant? Fine, at that particular time he will go to the water treating plant, complete the operation that's going on at that time, contact the shift supervisor. I'm finished; what's the next activity.

MR. ARNOLD: So basically they spend most of their time outside of the control room.

MR. STILLER: That's correct.

MR. ZUDANS: The shift supervisor is one of the SROs?

MR. TAYLOR: Yes, sir.

MR. ZUDANS: Where does the STA reside relative to the control room?

MR. TAYLOR: I think, as we were told yesterday, that the STA would normally work out of an office in the technical support center. He could come to the control room at any time.

MR. ZUDANS: So you could have a legal occupancy there of about four people, right, and then transients?

MR. TAYLOR: On a normal shift-work basis, there would be around four people here. There would be people in and out with job assignments, picking up procedures and so forth.

MR. LIPINSKI: Where there is a shift turnover, do the 14 people turn up in the control room to get their assignments?

MR. TAYLOR: No, sir.

MR. LIPINSKI: How do they know what their assignmens are?

MR. TAYLOR: The reactor operators will actually direct the people in the field under the direction of the shift supervisor.

MR. LIPINSKI: I am coming on shift, I have to report to somebody when I show up to my shift. Who do I show up to?

MR. TAYLOR: Normally, if you were reporting as a person who worked outside the control room, you would receive a turnover from your counterpart if there were a job in progress.

MR. LIPINSKI: How do I find him?

MR. TAYLOR: There would be a station I show up to.

MR. LIPINSKI: So when I show up to my shift, I go to some station in the plant?

MR. TAYLOR: Possibly. You do not necessarily go to the control room. There would be assigned stations for various outside personnel.

MR. LIPINSKI: Okay.

MR. ZUDANS: You have got to be a little more precise than that.

MR. STILLER: Mike, let me bail you out on that.

(Laughter.)

MR. STILLER: We have several categories of people. Let us define them. We have two supervisory personnel, senior reactor operators who go to the control room, are properly relieved, relieving their predecessor, and we have a shift turnover.

We have two licensed operators who go to the control room and relieve in the control room. We then have six personnel who are operating types of people outside of the control room. They would be showing up to set down their lunchbox, for one. They would be calling the control room, "I am here." Keep in mind they have no assigned spot, this is not "my desk," this is not "my workbench." They were directed there by their supervisor and relieved at that spot.

Now, there are four other types of people that we are looking at, other security now. This is the technicians, the I&C technician would go the I&C shop. The rad chem type of personnel would go to the chem lab or to the rad protection area.

So where he goes, really, it is a minor issue. It depends upon what type of job he is doing. Believe me, he must go somewhere, report to someone and be properly relieving. Procedures are set up for each and every one of them.

MR. LIPINSKI: Your explanation helps, because I could visualize where you are saying everybody departs to the control room where there are 28 people in that control room during a shift turnover.

MR. STILLER: Believe me, Mr. Lipinski, that is not the case. If a man is working, say, in a remote station in the water treating plant, for example, and it is 10:30 at night and his relief is not there, he is going to be calling, "Where is my relief?" This is a catalyst right there.

MR. LIPINSKI: That explains it. Thank you.

(Slide)

MR. TAYLOR: I will go on now and show some of the closer views of the portions of the control room and briefly describe what is located in these areas. As I said before, the turbine generator controls are located on this panel; the steam generator controls, main feedwater and auxiliary feedwater controls are located on this panel. Behind here we have associated turbine generator support as well as steam system and condensate and feed system equipment.

(Slide)

This is the reactor control station. Two video displays with the CRT keyboard, rod controls, nuclear instrumentation, rod control indication.

Then on the rear panel we have reactor coolant system indications and controls as well as cooling water, heating, ventilating, air conditioning. The two other visual displays.

MR. ZUDANS: The operator sits right here?

MR. TAYLOR: He would sit or stand in this area.

MR. ZUDANS: Whe he sits, he cannot see over the front panel.

MR. TAYLOR: Normally, when the operator sits, he would be on an elevated seat where he could see over it. You will notice this portion is slightly lower than even this one to aid in seeing over it.

MR. RAY: As I understand it, anything that is in storage in the CPU with the computer can be called up on any one of these four CRTs?

MR. TAYLOR: I would have to defer that question.

MR. MILTENBERGER: You have four CRTs in the control room with two sets of keyboards, one on the front panel and one on the back panel. You have the capability of the four CRTs that any of the information can be stored on any of the four CRTs. You can put the same pressurizer level diagram could be entered on all four CRTs simultaneously or anything that could be displayed on any one of the screens could be displayed on any of the others. So you do have that capability.

MR. RAY: The keyboards on all four, controlling all four, have access to all the memory in the CPU; that is what I am asking.

MR. MILTENBERGER: That is my understanding.

MR. RAY: There is no specific information that is limited to any one CRT?

MR. MILTENBERGER: That is correct.

MR. RAY: Thank you.

MR. WILKES: Richard Wilkes, Union Electric.

All the operator information would be accessible on the CRTs. Some programming type of information would only be accessible from the computer, but all the informaton that the operator needs --

MR. RAY: That is what I was interested in. Thank you.

MR. TAYLOR: The final portion of the front panel is the chemical and volume control system. Behind that is the safeguards equipment, electrical distribution within the power block. And then site-related electrical dstribution adn other systems.

(Slide)

MR. MARK: Safeguards equipment? This is like safety injection containment spray, RHR equipment such as that?

MR. RAY: Since TMI we have been concerned, particularly those who are concerned with human factors, with demarcation between systems on the panels. As I remember what I saw yesterday, this demarcation between systems is by way of color-coded tags.

MR. TAYLOR: Not necessarily, sir. Let me go back to the previous slide. We have no demarcation lines, as such. What we have is more of a functional group; systems are grouped within one area.

MR. ZUDANS: And you also had line mimic, which you do not show on this slide.

MR. TAYLOR: That was what I was going to show on the slide I was just one.

MR. MILTENBERGER: Mike, if I can address that, on the safeguards train, the nameplates that are affixed to the tags have different color coding for the different trains. I think this is what Mr. Ray is addressing. You might want to go into that aspect.

MR. TAYLOR: I will. This is an illustration of the mimicing that is used on the various systems. This is not used on all of the systems, but most of the major systems uses this, this being the chemical and volume control system.

We see here lines indicating piping systems with prompt control switches right in the mimic, showing the relative location of the pump in the system as well as valves and static components such as heat exchangers and tanks.

Also, we have the indications corresponding to those various parameters right above the control.

(Slide)

This is the safeguards system. As I pointed out on a previous slide, it shows the general layout of the rear board, with additional mimicing, portions of that system and indications directly above associated controls. I will address these panels on a close-up slide in just a moment.

(Slide)

This is a little better picture of some of the mimicing. It has flow arrows actually engraved on the little plastic begalite bars, and you can see how the various components tie right into the mimicing to show the locations.

(Slide)

This is the safeguards status panel, one of two, one for each train of safeguards systems. These panels will be used to alert the operator to both normal and abnormal situations.

First of all, there are amber lights that would come on, indicating the inability of a specific component to function. If, for instance, a breaker was open on a specific pump, the operator would receive an amber light, saying that pump is inoperable. That would warn him that there was a problem with that safeguards system.

Normally, this panel should be dark, no lights on for normal operations. Upon receipt of a safeguards actuation signal -- and I will use containment spray right up here (indicating) -- [illegible] portion of the status board should light up with white lights. The operator could then look and see if the individual white lights were on in the overall system, indicating that that system was performing the safeguards function.

In addition to this panel, the operator also has the indications on the control board to check.

MR. RAY: Let us assume something was blocked out of service, one of the auxiliary feedwater pumps, for instance, or main feedwater pumps. There would be tags on the controls on the bench board indicating they have been blocked. The status is there. So that if a man goes and reaches a switch that inhibits him doing something with it, he knows from the back tag the status of the system.

Now, when something is blocked out of service, is there a light in these positions on the panel, or are they only to indicate abnormalities in operation?

MR. TAYLOR: Depending upon the actual component and the indication used, like breaker control for pump motors such as that, we would get these lights under certain conditions.

MR. RAY: When equipment is in an abnormal state for maintenance reasons, you will have an indication here as well as the tag?

MR. TAYLOR: Yes, sir.

MR. RAY: So you will have a double check.

MR. TAYLOR: Yes, sir.

(Slide)

Two other status panels that I would like to point out. One is a rod position indication system. This utilizes LEDs, red LEDs that move up and down in relationship to the rod travel in the core. This provides a very quick and easy method for the reactor operator or other operator to determine the position of the rod upon a trip condition or something like that. He can easily look and see if all the rods have actually entered the core or not. This is actually from the position indication on the drive shafts themselves.

This is a trip status panel, indicating both the status of vario-- permissives interlocks in the control system and also the status of bistables in the trip and protection systems. So if we have a coincidence of two or four on some trip, we will get indication when just one of those four goes to the trip condition. So that the operator is aware of the status of the various systems.

This is for information such as during maintenance procedures or during actual instrument faults or abnormal conditions.

(Slide.)

Recently, we have completed three human engineering studies of our control board design. The first was a preliminary design assessment conducted by the Essex Corporation for the SNUPPS organization. The Essex Corporation reviewed both the Callaway simulator at Zion as well as the Callaway plant control board. Their findings were presented to the organization, and we are presently reviewing, resolving those findings.

Additionally, as was mentioned yesterday, the NRC completed a modified human factors evaluation. They were not able to evaluate all conditions such as lighting, heavy noise, furnishings, such as that. But a preliminary evaluation was conducted of the Callaway control room, and we are reviewing and attempting to resolve those findings.

One other study has been conducted. That was concerning the enunciator systems as well as the status panels, the safeguards status panels and the trip status panels that we just looked at. This was conducted by Westinghouse Offshore Power Systems and included both a review of the disp..ay information as well as establishing priorities of the various enunciator systems.

These three evaluations are being used to complete the review of the Calla..y control room and consideration for possible changes.

MR. ZUDANS: I have a question. Flash back the CRT picture, please.

(Slide)

I did not realize it was that far down. Okay. Good enough. On the CRTs the operator can recall functional diagrams of various systems. I was told he also can recall trends in certain parameters, a past value and a current value, showing the trend, say, reactor pressure, for example--

MR. TAYLOR: Yes.

MR. ZUDANS: -- show the curve showing how the pressure varies. I was told this trend could be recalled for quite a long time going back.

MR. TAYLOR: We can recall that trend also on paper recorders, such as this one right here.

MR. ZUDANS: I am more interested in the screen. Can the operator also recall your status diagrams here for six critical parameters?

MR. TAYLOR: On these displays?

MR. ZUDANS: Right.

MR. CERMACK: The displays have not been programmed yet, first of all; let me point that out. They have not been formally incorporated in the design. The displays would be current displays, not previous displays. It would give you a current indication on each display where you are. It is a snapshot in time.

MR. ZUDANS: I understand that. You have a primary coolant system recall which would show the current values of pressure and temperature at points. If they do change during the display, they will also change on your screen. It is a live display. I am now talking about the other part, the procedures part that takes care of those status trees. The question is whether you can call that up here.

MR. CERMACK: Yes, you can call them up, but they would be a snapshot of where you were at the point in time.

MR. ZUDANS: Would they be?

MR. CERMACK: Yes, they would.

MR. ZUDANS: That would be very interesting.

MR. CERMACK: But they will not give you the previous history.

MR. ZUDANS: I understand that.

MR. CERMACK: Which is what I thought was your question.

MR. ZUDANS: I am satisfied with the trending capability. That is another issue.

Now if the operator can recall the status trees and they are active snapshots for the current state, then we would see what is happening, what the results of these actions are.

MR. CERMACK: That is correct.

MR. CARBON: If I understand you correctly, though, Jim, it would not show on the screen. He would have to remember what it was 30 seconds before or 60 seconds. It would not show a path of what has been happening for several minutes.

MR. CERMACK: Right. There is no way to show on a snapshot status tree a path. You are either on one line or you are in another line. It is a snapshot of where you are at that point in time.

MR. LIPINSKI: Let me interject. It depends on how you want to write your software. One of the thoughts I had with respect to our trees is that they are time-dependent. As the accident is evolving, you can proceed along different branches.

As you point out, if you do not retain that information in memory, then you are going to recall it or you are going to rely on the operator, but if you chose to write the software properly, you could incorporate memory into those trees. It would depend upon how much you want to devote to memory storage inside the computer.

It is conceivable that you could use those trees to be time-dependent, not just snapshot.

MR. CERMACK: What you are saying is true. We had not planned to incorporate previous history and recall previous snapshots. That was not in our design.

MR. ZUDANS: Okay. Thank you. I think it is good. There is no point in dwelling on that anymore.

MR. WILKES: Richard Wilkes, Union Electric.

One feature that the computer system does have is the ability to take a moving time frame involving the alarms coming into the computer system. I do not remember the exact time frames. But if upon, say, a trip of the generator, the computer would have previous, say, five- or ten-minute history of the alarms that were coming in up to that, then it would record the alarms that took place after that, and they would be available on the memory such that the operator could go back and recover that information.

MR. ZUDANS: But I was distinctly told yesterday on the control room there was no connection between a CRT capability and the enunciators and all such things as you said they could not recall the past history of sequence of enunciators on the CRTs.

MR. WILKES: There is a sequence-of-event monitor within the computer.

MR. LIPINSKI: What you have just described is what happened at TMI, where the computer was just overloaded with alarms and the operators were not able to keep up with that volume of information. The computer could not keep up with it.

It was not the computer system but that amount of information did not help them one bit with the volume that came in because it listed the important information in with the miscellaneous information and it all got lost in one big jumble.

MR. WILKES: I understand what you are saying. And that is one reason why this is recorded on the magnetic memory as opposed to being printed out in the control room: to reduce the amount of information which the operator is receiving.

MR. CERMACK: Let me clarify the status trees again, what I said about the snapshots at any given point in time. However, we do have these trend graphs that were previously referred to by Mr. Taylor, where you can call up, for example, the reactor coolant system pressure as a function of time, you can call up hot-leg temperature as a function of time, the parameters that you are measuring, reactor vessel water level. You also have a trend graph. So the parameters you are measuring you can call up as a function of time.

MR. CARBON: Are there further questions?

(No response.)

Are you implying that that finishes your presentation?

MR. TAYLOR: Yes, sir.

MR. CARBON: I guess we have no more. Thank you.

MR. PASSWATER: I have to introduce myself for this particular topic.

(Laughter.)

MR. PASSWATER: The purpose of this presentation is to go through an item that has been of interest to the ACRS, to the NRC staff in the most recent applications. The generic title of the topic is systems interaction. I would like to do two things, the first one being to describ briefly the background of te topic and what we understand the major points of the topic are.

Secondly, I would like to discuss what in the SNUPPS design we have done to address the concerns of systems interactions.

(Slide)

I would also like to say as kind if a sideline that systems interaction responses I am going to address include the effects of internal flooding. Internal flooding was culled out as a separate agenda item in one of the earlier agendas we received.

By way of background, the systems interaction came about as a result of some operating experience from operating plants. The concern that was perceived by the staff was that things were happening in these operating plants that were unacceptable or undesirable, and that the probable cause of that was that there was lack of any interdisciplinary review during the design of those plants.

In other words, the architect-engineers had their civil group, they did their thing, the electrical group did their thing, the mechanical group did their thing. Nobody talked to anybody, and they all came out with a powerplant. But they had not really considered the possible interaction of different systems and different hazards. So it is now carried as an unresolved safety issue, specifically A-17, and it is also a topic of the TMI Action Plan NUREG-0660 as Item 2.C.3.

To date the action that has been taken on this topic was there was a study done by Sandia on a typical LWR. It was categorized as a limited study on systems interaction. They identified some action items which had been addressed by the staff, and I believe that the TMI Action Plan was recommended by the staff, by the Commission, that further work needed to be done in this area generically.

(Slide)

That is the background. The next part of my presentation deals with the SNUPPS approach and what we have done in regard to the area of systems interaction.

I would to say that the design of the SNUPPS took into account in a broad sort of way the area of systems interaction from day one. Yesterday on your tour of the plant one of the things that we tried to point out on the tour was the concept, the separation concept that we used in the design of the plant.

There is an upper and lower cable spreading room, redundant trains are -- come into the control room through two separate cable spreading rooms. We have two separate electrical penetration rooms and two separate mechanical or piping penetration rooms. Those are located at a considerable distance -- there is a considerable distance between the redundant penetration rooms around the containment.

So the systems interaction idea specifically with regard to separation was an initial consideration in the design of Callaway SNUPPS design.

There have been some additional studies done with regard to certain aspects of the systems interaction. Those are the hazards analysis that we have done, FSAR Appendix B. There was an item that came up during the FSAR review on control system failures, which is addressed in the questions and answers in the FSAR.

There was a separate question in the envirnomental impacts on the control systems which was also documented and answered in the FSAR. And then a study that is now underway is one on heavy loads analysis, which is again a separate category of hazard.

(Slide)

I am going to go through the basis for the hazards analysis. Again, I am not planning to get into any detail on the hazards analysis itself. It is very lengthy. As I mentioned, it is in Appendix B in the FSAR.

It considers four types of hazards. Those are: pipe break effects on systems; missiles both internal and external; earthquake-induced failures, specifically nonseismically designed systems; and fire hazards.

Some of the major assumptions -- these are not the only two assumptions, but they are kind of key to the way the analysis was done -- first of all, it is assumed the plant is operated by the technical specifications. The reason that was looked at is that would delineate times, limiting conditions for operation and so forth. The second one was we did not assume a loss of off-site power except for cases where the hazard itself would result in the turbine trip or generator trip or reactor trip.

The first of the types of hazards we considered, pipe break effects. Pipe break effects includes flooding, which is one of your topics that was culled out separately earlier, jet impingement for pressurized line, pipe whip, and environmental effects.

The analysis that has been done on pipe break is in accordance with the NRC branch technical positions MEB 3-1 and ASB 3-1 criteria. Those criteria dictate where breaks are assumed for moderate high-energy lines and als dictate what happens to the piping system following a break. Those are again high- and moderate-energy lines.

We also did a study because it was a specific licensing concern back at the PSAR stage. That was in the rupture of circulating water expansion joint and the consequences of that rupture flooding from the cooling tower into the turbine building.

The second type of hazard is missiles. Missile analysis, missile hazards analysis is specifically culled out in the FSAR section 3.5. Internal missiles consist of rotating and pressurized component sources. Selecting their trajectories and so forth is culled out in 3.5 of the FSAR.

External missiles, we considered tornado missiles and the turbine missiles. As Mr. Schnell pointed out yesterday, the plant is arranged in a peninsular fashion. The expected high-energy missiles from a turbine failure would not be perpendicular to the reactor building or the rest of the plant.

Tornado missiles were analyzed in accordance with the NRC regulatory guide. Essentially, we say that the tornado missiles are external because the plant is designed to exclude those missiles from entering the plant and coming in contact with any safety-related equipment.

The next category of hazard was earthquake-induced hazards. This was specifically addressed in Reg Guide 1.29; specifically, position C.2. That position states, essentially, that -- it lists categories of safety-related equipment -- and essentially, it says that nonseismically designed euqipment which is not designed to function following an earthquake should be restrained or should not be allowed to affect the operation of the seismically designed safety-related equipment that does have to function following an earthquake.

This is a very great portion of the hazards analysis that has been done on SNUPPS. Hazards from the earthquake-induced failure include drop impact forces and spray and flooding by an outside designed piping system.

I should point out that resolution on seismically induced failures as well as the other hazards we have talked about has been by prevention of the hazard occurring rather than by analysis of the effects of the hazards.

In other words, if we have a nonseismically designed line that passes through a space where safety-related equipment is located, rather than analyze the effects of that line breaking on safety-related equipment, we have installed what we call two-over-one piping restraints to prevent the line, the breakage of that line from impacting safety-related equipment.

We have also done a very extensive fire hazards analysis.

MR. LIPINSKI: Before you proceed, on the earthquake-induced failures can you comment on your cinderblock walls and if the hangers and the effect on seismically qualifed equipment?

MR. PASSWATER: Masonry walls at Callaway are all seismically designed. We do not have any nonseismic in safety-related areas, that is.

MR. LIPINSKI: So you did not have an interaction in that area?

MR. PASSWATER: We did not have that interaction; that is correct.

We have also done an extensive fire hazards analysis, which is in the FSAR section 9.5 and the four different appendices to that section. Essentially, what was done there was we defined fire areas. They are all given a number which shows that they are separate fire areas. Those areas are separated by three-hour or greater barriers, again talking about safety-related areas.

And the analysis that has been done on that is to look at the equipment that is located in the spaces and the effects of not only the fire but also the subsequent fire suppression system actuation and the capability to safely shut down the fire, given the fire and given the fire area.

MR. RAY: I assume that these analyses for these various hazards are based on installation as designed rather than how they were installed, the various facilities of the various systems. Do you anticipate or plan any walkthrough audit of the plant at each stage of construction to make sure that actual installation has not introduced some hazard that you did not anticipate in your design?

MR. PASSWATER: That is a good point. I forgot to bring it up. The actual hazards analysis was done using the three-quarter-inch scale model of the SNUPPS plant which is located in Bechtel's office in Gaithersburg, Maryland.

The model is somewhat unique from other models of large powerplants in that it includes all sizes of lines; for example, even down to instrument lines. There is very little at Callaway that is field-run. Everything was designed by Bechtel in Gaithersburg and is incorporated in the model. The actual hazards analysis was done on the model looking at, in plastic, what exists in the field at

Callaway.

As far as plans for a walkdown to verify what we have done in the hazards analysis, as of right now I do not believe there are any specific plans to do that kind of walkdown. There are other design verifications that are done in the field, such as walkdowns, but as far as I know, none of those specifically address what was done in the hazards analysis.

MR. ZUDANS: You said a three-quarter-inch model?

MR. PASSWATER: Three-quarter-inch equals one foot.

MR. ZUDANS: And that is sitting at Bechtel?

MR. PASSWATER: It is in a room a little smaller than this.

MR. ZUDANS: And you say all the piping is represented?

MR. PASSWATER: That is right.

MR. ZUDANS: But nevertheless, you will have to verify that your drawings represent the actual drawings as built.

MR. SMITH: Joel Smith, Bechtel.

When we evaluated nonconformance such things as large pipe, small pipe, we take that into consideration. And we would not disposition one that would affect our two-over-one analysis.

MR. CAPONE: Don Capone, Union Electric.

I would like to clarify this a little bit, the two-over-one. Most of the activities on the quality control program are under a Q program quality control check. Any discrepancies from the drawings have to be dispositioned by the designer.

Two-over-one piping systems that we talk about are under a limited Q program, which involves checks by quality control people to verify their installed as shown on the drawing. If they are not, they have to be referred back to the designer for disposition.

I do not know if that addresses your concern, but it is under the quality control program. That is how we verify that it is installed the way it is shown on the drawing.

MR. ZUDANS: In other words, if they would find --

MR. CAPONE: There is no field engineering, I guess is what I am saying, it is designed back in Gaithersburg. If it is not installed that way, it is under the quality assurance program, it will be a nonconformance, it will have to go back to the designer to be dispositioned. And that evaluation would be documented.

In other words, if it is not where it is shown on the drawings, it has to be verified that it is acceptable to be left there or it has to be moved back to where it was originally shown.

MR. ZUDANS: Let us walk through an example. Yesterday we saw a drawing with the piping run up there. Now, he goes and takes a drawing and puts that pipe in place. It is not likely that he would put it exactly within an eighth of an inch of these things, is it?

MR. CAPONE: There are tolerances.

MR. ZUDANS: Does he look at that after he has done it, look at the elbows and what not, and compares it with the drawing?

MR. CAPONE: There are two things when you talk about the drawing. Yes, if it is a safety-related system, a two-over-one system that is under the quality control program, then there would be a separate check after it is installed to verify location.

In addition, there is a 79-14 walkdown of the piping systems that is going to come later. That is an IE Bulletin that requires a complete walkdown for location of hanger snubbers as a final check.

Sp when we speak of piping, you might say there is a double check on the cable tray and that it would be under the quality assurance program. It would be checked out at the final location. There is no final walkdown on cable tray like there is in the piping system.

MR. ZUDANS: But it would not be then for safety-related piping?

MR. CAPONE: If it was a two-over-one system, it would be.

MR. ZUDANS: Could you explain the two-over-one?

MR. CAPONE: That means you have got a seismic Category I system, but you have got a system that is not a safety system, but it is hung over the top of that system and you have to design systems to assure that that is not going to fall down or in any way impact the safety system.

So our way around that was to design the supports for those systems so that they would not fall down, rather than to evaluate the hazard of it falling.

MR. ZUDANS: So what you are saying, your hazards analysis would be based on actual installation for all safety-related systems and two-over-one systems as defined by you now?

MR. CAPONE: Yes.

MR. ZUDANS: You do not particularly worry about deviations of a foot or inch in systems that do not have an impact on anything that is safety-related?

MR. CAPONE: That is right. And there are some tolerances in the specifications. There maybe tolerances up to two inches; in other words, that pipe could be within two inches or an inch and a half, depending on that system and how critical it is. Those tolerances are looked at when they do the analysis, that if that pipe happens to be an inch and a half and is impacted.

MR. PASSWATER: I might point out also that the way the analysis is done, if it is close, if it is possible that the tolerance is going to put it close, it has been restrained; that has been the philosophy.

MR. ZUDANS: Your piping system is designed and analyzed before it is built. The analyst assumes that it has certain space configurations. He determines where supports go. Supposing you did not do what you are saying you are doing now. You did not install it exactly; you relocated the supports a little bit because of convenience. All of the analysis -- it does not mean the piping system is not good, but the analysis would not be of system as built.

MR. PASSWATER: If it was not built as designed, that would generate a nonconformance report from the field which would go backl to the designer for disposition. One of his things that he does with NCRs is look at the effect of that on hazards analysis, by the hazards analysis at Bechtel.

MR. ZUDANS: Thank you.

MR. CAPONE: In addition, you have got the 79-14 walkdown, which is a requirement.

MR. ZUDANS: That is where that came from.

MR. CARBON: When you make the design change, do you make the same change in this big mockup at Gaithersburg?

MR. CAPONE: Yes, sir. There are two things that happen. We talked about the standardized concept. If there is a change that is made at Callaway, it is made at Wolf Creek and it is made at the model.

MR. RAY: Are these nonconformance reports required only for safety systems and two-over--one systems, ore are they required also for nonsafety systems?

MR. CAPONE: For all systems.

MR. RAY: Do I understand clearly that you have no site design of any of the systems?

MR. CAPONE: That is correct.

MR. PASSWATER: Inside the SNUPPS power block we are talking about we do have a site AE.

MR. CAPONE: None of that work is done on site. None of that design work is done on the site.

MR. RAY: So there are no field innovations then that are likely to occur that will incur interactions that were not anticipated in your hazards analyses?

MR. CAPONE: That is correct.

MR. RAY: How about the problem of an EMI interaction, electromagnetic interaction induction, is this given any specific consideration from interactions anticipation, or do you rely on the original design to have anticipated that and avoided it?

LR. GASDA: I believe what you are referring to is interferences particularly with instrumentation systems?

MR. RAY: Yes. False signals developed because of heavy current induction between wiring.

MR. GASDA: The way we have designed our cable tray and conduit system is that we provide for spatial separation between what we identify as instrumentation cables, low-level signal cables, more than 4 to 20 milliamp or thermocouple cables such that we maintain a spatial separation between those cables and large power cables.

One of the things we do, to start out, we have 13 20-KV power cables on the top layers, 41 60-volt cables on the next layers, 600-volt power control and then instrumentation cable. So instrumentation cable is always maintained separately from the cables which interfere.

MR. RAY: Now, once again the wiring systems, power signals and so on, are subject to nonconformance reports also. So if anyone should violate those standards, yoy would know it.

MR. GASDA: That is right. We provide for specific routing of cables through the tray system. That is controlled by the quality control program and our quality control program such that any deviation from that routing is immediately brought back to us for our evaluation.

MR. RAY: And construction QC surveills this, I presume?

MR. GASDA: Yes.

MR. RAY: Nonsafety as well as safety systems?

MR. GASDA: Yes.

MR. RAY: Thank you.

MR. CAPONE: Yes.

MR. ZUDANS: Could you tell us a little bit about that computerized system that you have for cable tray numbers and cable routing? It would appear that it would be impossible to route it incorrectly from what we heard at the plant.

MR. GASDA: A qualified "Yes." It is impossible to route it incorrectly, although mistakes are made and this is what the nonconformance program is designed to pick up.

MR. PASSWATER: I just wanted to finish up on hazards analysis by saying that it is an interdisciplinary review. There is a task force made up of the different disciplines on the project. It is an ongoing review. It is not a static thing that has been completed; it is ongoing. It takes into account these kinds of changes we were just talking about.

MR. LIPINSKI: On your assumptions you list technical specifications. Licensee event reports are still being written at the same rate with no change in violation of the tech spec. It initiates the licensee event report. Your conclusions are based on this assumption.

Would you change your conclusions if you did have a violation of the tech spech on some portion of the system that you are reviewing? How sensitive is your assumption or your analysis to this initial assumption that you are performing going to the tech spec?

MR. PASSWATER: It is not very sensitive at all to that. If we are talking about a case where I have a deviation from a technical specification that would generate an LER, I would have to have, in order to be outside the analysis that I have done, I would have to have this particular hazard occur at the same instant I was outside the technical specifications. So these two things would have to be concurrent in order for me to be in the position that was not analyzed by the hazards analysis.

MR. CARBON: Mr. Stright.

MR. STRIGHT: Just one further addition to what Mr. Passwater said. I do not believe either of those assumptions are particularly pertinent, because of the conservative nature and the way that interactions are corrected.

For example, the seismically induced failures, as I said, if there is any remote chance that something could fal or damage or break a safety system, rather than doing a complicated analysis in order to determine whether this really would fall over here, we fix it. Therefore, the assumptions of the technical specifications and the loss of off-site power are not really critical and limiting.

MR. ZUDANS: In other words, if you find a hazard exists, you simply eliminate the source of it?

MR. STRIGHT: Yes, we fix it rather than analyze our way out of it.

(Slide)

MR. PASSWATER: Three other specialized analyses that we are doing -- I should say of the four I am talking about, hazards analysis is certainly the biggest and the most complex, the most man-hours spent and so forth, the one we just talked about.

Control systems failures was specifically an NRC question that came up during the FSAR review. In response to that question we did an analysis documented in the FSAR. The concern was do we have multiple control systems failures that could occur from a single event that would put us outside what we have analyzed for Chapter 15 of the FSAR?

So we looked at the control systems, loss of a single instrument, break of a single instrument line, and loss of a power supply where that power supply supplies power for more than one system and more than one component

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in the control room. It was done for the five major NSSS control systems: reactor control, steam dump system, pressurizer pressure control, level control, and feedwater control. That analysis is, as I said, in the FSAR.

The conclusion of it was that any of those occurrences on any of those systems were all bounded by what we call Condition II events in the FSAR. Those events specifically are loss of feedwater event or turbine trip event.

(Slide)

Another analysis that was done, again in response to an NRC question, is the environmental impacts and control systems, on control systems. These were initially identified by Westinghouse, I believe, in a report to the Commission about certain conditions in a plant, such as the small steam break or a moderate steam break which could impact control systems that might interact or might affect the operation of safety-related systems.

Two of those on the power-operated relief valves on both the primary and the secondary side were not applicable to SNUPPS because we have qualified those PORVs to the environment that was discussed in that report by Westinghouse.

Two others, the impact on rod control system by break in the turbine building and the other one had to do with the steam generator reference leg, had already been answered in other questions to the NRC. We talked about steam generator reference leg yesterday and the fact that may be resolved very soon.

The turbine instrumentation was actually -- we were looking at qualifying the instrumentation for that particular one. So it will be in the same category as the two PORV issues.

(Slide)

The final one I am going to talk about is the heavy loads analysis. This was brought about by a generic letter from the NRC staff to all the licensees to look at, procedures and administrative controls on lifting the cranes currently installed and temporarily installed in the plant, to look at the possible effects of dropping loads.

We had a study underway. We have made a preliminary response to the NRC back about six months ago. The results of the completion of that study are expected to be done sometime before the end of this year. And that will be reported to the staff also.

Are there any questions?

MR. ZUDANS: Yes, I have a question. It relates to other interactions not discussed by you. You have the RHR system which is designed for lower pressure and reactor coolant designed for higher pressure. The two systems are separated by a couple of valves. What do you do with the dead space between these valves? Do you monitor it?

MR. PASSWATER: Monitor it for?

MR. ZUDANS: Whatever. Whatever you do, I would like to make sure that that interface between these two systems is not violated.

MR. PASSWATER: John Prebula from Bechtel.

MR. PREBULA: We have a perfectly installed test system that tests the leakage rates through each valve. And periodically, we test the flow rate through the valves. Downstream of the valve there is a relief valve which relieves the pressure from the relief valve if there is leakage past the second.

MR. ZUDANS: How can you test both of these valves? You do not know where the leakage comes from.

MR. PREBULA: We test each valve individually with this remote testing system. It is permanently piped up through proper valve configurations. We measure the flow from each valve individually.

MR. ZUDANS: Yes?

MR. PREBULA: We test that the valves are actually closed during normal power operation as you are coming up in pressure. And then we test the leakage rate from the valves.

MR. ZUDANS: Okay, let us leave that.

There is another system I would like to hear about. You have an instrument and a service air system. Are these interconnected in any way?

MR. PASSWATER: This is Dennis Grove from Bechtel.

MR. GROVE: The air service is tied into the air service system. There is an automatic isolation valve that isolates the instrument air from loss of flow. They are inter-tied.

MR. ZUDANS: What is used for separation of these two systems? How easy is it to open one and get air from the other?

MR. GROVE: I will have to get out the pipe drawing and get back to you.

MR. ZUDANS: That is all right.

MR. LIPINSKI: Do your demineralizers use any flushing operation to break up the resin beds wih air and water?

MR. SCHWOERER: Frank Schwoerer of the SNUPPS staff. We have deep bed condensate demineralizers, and we do have a resin regeneration cycle that we go through in which we flush the resins out of the demineralizer vessels and take them off to other vessels where they are treated.

MR. LIPINSKI: Are you using air-water combination in these operations?

MR. SCHWOWERER: I do not believe we are using air.

MR. LIPINSKI: Because one of the things that happened at TMI, the early initiating part of that accident, is that they drove the water into their service air system and that was interconnected to the instrument air system, and the water got into the instrument air.

MR. SCHWOERER: Three Mile Island has a different type of demineralizer. It is a Powdex, P-o-w-d-e-x, type demineralizer. We have a different type. It is a large vessel containing resin beads, so-called deep-bed demineralizer. And the method of regenerating these beads is totally different than what was done at Three Mile Island.

MR. CARBON: Any other questions?

MR. ARNOLD: I would like to ask a question. If it takes a very involved answer, I will drop it. But when you look at systems interaction, generally you are looking at interactions between hardware or between a person and hardware. Superimposed upon that is a person-to-person interacting system.

Do you stop to look at that to see whether it or not it is appropriate for handling the particular problem and whether or not some other interaction of person to person prior to -- well, in the design phase on, really, might have reduced the probability of the particular undesired action?

MR. PASSWATER: In the design stage, as mentioned, the hazards analysis, for example, is an ongoing interdisciplinary review. It is several people in the hazards analysis task force at Bechtel that are looking at not only their own systems but as a group looking at the system and looking at potential hazards from the design standpoint.

Looking at operation of the plant, the procedures that we talked about earlier, both emergency operating and normal operating procedures Mr. Schwoerer talked about, and the training, the extensive training we give operators, to point out that those potential pitfalls and unsatisfactory interactions, the training does incorporate that. So the operators from the time the plant loads fuel on, are aware of that. They are retrained, and their memory is refreshed on those areas.

MR. ARNOLD: My question was prompted by the fact that earlier in the week in the Wall Street Journal there was an advertisement of a well-known engineering firm -- the name escapes me at the moment -- offering person-to-person systems interaction analyses.

I thought there might be there some thought that not the group that is concerned with the design and so on, but the people who actually get involved in communicating about the particular incident that happened or who had some responsibility for communicating on the procedures relative to the carrying out of the procedures rather than necessarily the development of the procedures.

MR. PASSWATER: We talked yesterday about the various review safety committees that exist to support the operation of the plant. One of their functions is, for the on site review committee and the NSRB, is just such a thing

-- reviewing abnormal occurrences, what went wrong. It is also an interdisciplinary or interdepartmental-type review process.

MR. ARNOLD: Thank you.

MR. RAY: John, is it possible that the Wall Street Journal ad was an ad for marriage counseling?

(Laughter.)

MR. ARNOLD: Well, I wouldn't have noticed it if it was.

MR. CARBON: With that let's take a short break.

(Brief recess.)

MR. CARBON: Let's reconvene.

MR. PASSWATER: The next presentation is on hydrogen control, and this was called out in the agenda as being something by the staff and the Applicant.

Dr. Carbon, I don't know how you want to handle that. We have a presentation that we can give you now or later.

MR. CARBON: Do you have many comments?

MR. EDISON: I have just a few.

MR. CARBON: Why don't you lead off?

MR. EDISON: Gordon Edison, NRC staff.

(Slide.)

I'm going to present the staff position on hydrogen control for large dry containments and relate it to the Callaway-1 review.

(Slide.)

First, I will talk about the staff position, and second, I will present a few details about the Callaway design and show how that relates to the staff position.

(Slide.)

The staff position on the review process at this time is essentially the pre-TMI requirement; that is, 10 CFR 50.44 for core response for something like five percent metal-water reaction or equivalent to approximately five times the amount of hydrogen calculated in the ECCS calculation.

That doesn't mean that the staff is ignoring the effects of the Three Mile Island accident. I will get to that point.

Let me mention the second bullet here; that is, the staff position on Callaway and review of Callaway is consistent with the full power licensing of North Anna-2, Salem-2, Farley-2, and the TMI restart review. The staff does have a proposed interim rule which is not now imposed on large dry containment designs. That proposed interim rule is continuing to be revised. It is in draft form, and the staff is not now imposing that rule on large dry containments.

The staff does have confidence that large dry containments can withstand hydrogen generation in an accident, so the staff is not overly concerned that we have a problem here. We expect that the interim rule that is prepared and developed for hydrogen control is not going to be difficult for large dry containments to meet.

Are there questions in this area before I go to the Callaway plant specific review?

(No response.)

(Slide.)

The Callaway plant, the containment has two and a half million cubic feet of free volume. The staff has looked at a TMI-2 type event which we estimate to have about a 30 to 50 percent metal-water reaction, and calculated that that gives about a 6 to 10 percent hydrogen uniformly mixed concentration in the containment, which is well below the detonation limit.

The design pressure of that containment is 60 psi, and failure pressure is twice the design pressure. It could be larger than 120 psi. It probably would be smaller than three times the failure pressure.

The staff feels the design can accommodate pressure from hydrogen combustion. We have an experimental data point at TMI where there was a large dry containment. The pressure didn't approach the 120 psi or anything like it.

In addition, the staff has laid on a whole bevy of post-TMI requirements, so we feel that type of accident is considerably less likely to occur in the future than it was in the past.

You gentlemen are aware of NUREG-0737 requirements the staff is laying on new applicants at this time to examine their auxiliary feedwater systems and improve the reliability of those. We've made operator training improvements in programs to require things like safety valve and PORV valve qualification and training, and many more.

So what we think we have here is an accident that is less likely than before. The large dry containment has a large amount of volume to contain hydrogen should it be generated, and we don't think there will be a problem there. This is an unresolved safety issue within the Commission. The large dry containments I think have less of a problem.

MR. LIPINSKI: On your 6 to 10 percent hydrogen you're assuming perfect mixing is in that volume.

MR. EDISON: That is correct.

MR. LIPINSKI: You've discounted the possibility that there's going to be stratification, and the hydrogen will rise and collect in the top of the containment?

MR. EDISON: No, the staff has not overlooked that possibility. You have to look at pocketing. Of course, if you have pocketing you will have smaller -- possibly a smaller local combustion.

The staff is reviewing this. However, the staff does feel that the containment is large enough, has enough volume that it can withstand this. We can't get too much into the technical concerns about bringing in some of our reviewers, and we will have those people available at the full committee meeting next week.

But some of the scenario kinds of things that go on here are that if you do have a metal-water reaction, you're also going to have steam generated in the containment to help dilute the hydrogen and keep it down below the detonation limit.

At a later point in time after say a LOCA type of accident, the steam will be condensing and leave you more pressure capability for the containment to accommodate a large amount of hydrogen.

MR. LIPINSKI: But if you go back to the TMI-type of accident where the vents are now being provided on the primary system to get the gas out, their venting was to shoot directly upward to the top of the containment so it would give it a velocity stream upwards, and no provision to direct the hydrogen to the collectors that will remove it from the containment.

The natural path is to rise and possibly pocket in the very top. The ventilation system simply stops at the crane rail level, and there's nothing in the very top of the containment.

MR. ZUDANS: There is something I forgot to ask. I saw an outlet right at the top of the roof. What is that in your containment? Is it inlet or outlet ventilation? Right on the top of the dome.

MR. CAPONE: Are you speaking of the duct that comes down?

MR. ZUDANS: From the very top of the dome.

MR. LIPINSKI: It looked like a cage.

MR. CAPONE: I think that is temporary.

MR. PREBULA: That is the plant unit vent. That is an elevated release point for the buildings. It is not associated with the containment structure itself. It originates in the auxiliary and fuel buildings.

MR. ZUDANS: It's not the vent outlet for the containment?

MR. PREBULA: Yes, it is, but the containment purge systems enter that system and then get exhausted up the side of the containment and out.

MR. ZUDANS: Okay. But that is one of the purge points. It would be the inlet for your purge system and go out some place through the wall.

MR. PREBULA: No. That is -- that's not the inlet to the purge system. It is the discharge from the plant.

The purge system takes suction from the auxiliary building roof.

MR. LIPINSKI: Is there a hole in the very top of the containment?

MR. PREBULA: No, sir, there is not.

MR. SCHNELL: Don Schnell, Union Electric.

Don, I think he probably saw what was an opening, an access opening intended for construction.

MR. ZUDANS: No. It's a big duct.

MR. SCHNELL: The duct you saw on the outside of containment --

MR. ZUDANS: Inside.

MR. LIPINSKI: It's a cage that rises from the crane rail level. It follows the contour of the containment right to the center of the top of the containment. It looks like a cage-mounted duct.

MR. SCHNELL: Oh, I know what you're talking about now. There is what we call a maintenance truss that takes that configuration. The access is from the crane elevation. And the idea of that is to service the fixtures that are in the interior of the dome.

MR. LIPINSKI: It has nothing to do with ventilation.

MR. SCHNELL: No.

MR. CARBON: Other questions?

MR. ZUDANS: Yes. I have one question. The proposed rule that's being revised, what level of water-metal reaction is in that one?

MR. EDISON: It's changing weekly.

MR. ZUDANS: Moving target.

MR. EDISON: The numbers have been bouncing around. I don't think really the technical people who are going to set some number have the large dry containments in mind. I think they're more concerned about small containment designs.

This isn't to say that the staff wouldn't impose the interim rule also on large dry designs. It's just that at this time the staff feels the large dry containment designs are much more likely to be able to deal with the interim rule without fixes. So we have a fairly high confidence that the large dry containment designs will not have a problem even with the proposed interim rule, however it develops.

MR. ZUDANS: Well, of course I agree with you. The only problem that bugs me is this. If you set the source of the hydrogen for one plant at one level and for a different type plant at a different level, it doesn't quite make sense to me unless there are some specific features in the plant that really --

MR. EDISON: I'm not saying the rule will be different for different designs.

MR. ZUDANS: I know some plants are required to take care of one hundred percent of metal-water reaction. They have to do two design systems to cope with that. This is 30-50. I've heard 75 percent. So it's really a moving target.

MR. EDISON: That's right. This is -- this 30 to 50 is not a staff position. The staff simply looked at what happened at Three Mile Island and said what would happen at Callaway.

MR. ZUDANS: But you do accept this for the Callaway design.

MR. EDISON: No. The 30 to 50 is not a design requirement for Callaway. Five percent, as I showed on my first slide, 50.44 is the requirement.

MR. CARBON: We can move on now.

MR. PASSWATER: Jim Cermak will --

MR. EDISON: There was one more thing in my presentation that I have to show you. This is a prepared staff statement from the review branch on hydrogen control.

(Slide.)

I will read it for those in the back who can't see it up here. It's in the handout. It says, "Pending the rulemakings called for in item II.B.8 of the TMI Action Plan, NUREG-0660, the staff is not now imposing any new requirements beyond 10 CFR 50.44 for hydrogen control in large dry containments. The staff completed its reviews for the full power licensing of North Anna-2, Salem-2, Farley-2, and TMI-1 restart without imposing any new requirement for hydrogen control. This decision was based on the staff's view that large dry containments had sufficient capability to accommodate hydrogen combustion so that no new requirements needed to be imposed pending the rulemaking proceeding on degraded core cooling. In the proposed rule portion of the interim rule, which is documented in SECY-81.245A, the staff proposes to require that certain specific analyses be performed for all large dry containments to assure that containment integrity and safe shutdown will not be jeopardized by hydrogen released from the postulated degraded core accidents."

MR. CARBON: I would like to go back to Dr. Zudans' question. You are imposing different requirements on different plants in terms of the amount of hydrogen being released, are you not?

MR. EDISON: Do you want to talk about large dry containments?

MR. CARBON: I'm just saying across the board irrespective of containment, you're saying in some plants you assume so much hydrogen and other plants you assume some different amount.

MR. EDISON: No, I'm not saying that, and I'm not sure I could tell you what is happening on large dry containment designs. I am prepared here today to deal with the agenda questions on large dry containment.

MR. CARBON: Would you be prepared to answer that question next week, tell us how this varies from one to another?

MR. EDISON: Certainly.

MR. LIPINSKI: Going back to your assumption of perfect mixing, the one data point that's available is TMI-2 where hydrogen was vented within the containment and there was a burn.

Has a conclusion been drawn based on some of the investigations as to where that hydrogen may have been pocketed?

MR. EDISON: I've heard different opinions expressed. I don't think a conclusion has been drawn and probably won't be drawn until people go inside the containment and look.

The number that I remember from the pressure generation is a peak of something like 28 or 29 psi, which is well below the failure pressure here -- anywhere from 120 to 180. But as to where it happened or where exactly it occurred, I don't know.

MR. LIPINSKI: But I think it relates to this assumption of perfect mixing.

MR. EDISON: Yes. There are a lot of technical parameter considerations in trying to come up with the proposed rule, obviously, and there's a lot of manpower and thinking going into it.

MR. ZUDANS: There are lots of experiments being done right now.

MR. EDISON: It's not something that -- we could spend hours developing that question here if we wanted to. I don't think we want to do that. But we will have the staff, who can talk to the position on other containments, with us at the full committee meeting next week at your request.

MR. CARBON: Let's move on then.

MR. EDISON: That completes the staff's presentation.

(Slide.)

MR. CERMAK: Jim Cermak, SNUPPS staff.

Let me briefly describe the hydrogen control system that we have on the plant and the assumptions that went into the design basis for the plant, and then I will move on to other scenarios.

The hydrogen control system consists of redundant hydrogen recombiners, a redundant hydrogen mixing system, redundant hydrogen monitoring system which uses thermoclimativity to analyze hydrogen concentration, and it has a backup hydrogen purge subsystem.

MR. MARK: The particular hydrogen monitoring system, how long does it take to get a reading from it and with roughly what accura y does it perform?

MR. GROVE: Dennis Grove, Bechtel.

The hydrogen monitor we have has the capability to be put in the standby mode when the heat tracer is on and the generator within the cabinet itself stays hot. The time it would take to get a measurement would just be the sample transport time from the time you open the isolation valve and pump the sample to the analyzer. I don't have that right now. I can get that for you.

MR. MARK: It's not important, but within a percent or something?

MR. GROVE: It will be something like that.

MR. CEIMAK: The sources of hydrogen used in the design basis analysis consisted of a five percent zirconium-water reaction post-LOCA. This, I should point out, was about 50 times the calculated core-metal reaction for the worst case loss of coolant accident.

(Slide.)

The post-LOCA radiolytic decomposition of water and the post-LOCA corrosion of metals and paints.

(Slide.)

This was all combined together as a function of time during the accident to look at the hydrogen concentration in the containment as a function of time after the loss of coolant accident.

In this particular curve it assumed that the hydrogen recombiners came into operation one day after the loss of coolant accident. As you can see, the hydrogen volume concentration never exceeded two percent. Then it came down as a function of time after the accident as a function of the hydrogen recombiners.

MR. LIPINSKI: This is under a perfect mixing assumption?

MR. CERMAK: This is under a perfect mixing assumption, that's correct.

(Slide.)

But these are just hydrogen recombiners. We are well below the flammability limit for hydrogen even in an atmosphere without steam at all, because the flammability is about four percent.

MR. LIPINSKI: Your two percent is based on your total volume of containment, but if I restrict that hydrogen into a smaller volume, the concentration goes up.

MR. CERMAK: Yes, it is spread out over all the containment, but there are studies that show -- for example, Bechtel-Frankfurt -- that show that you get very good mixing. And let me hit mixing right now since it seems to be your interest.

When you have a loss of coolant accident you have a release of steam-water mixture, and you have a lot of steam in the containment. And when you -- first of all, you create a lot of turbulence. Second, when the sprays come on, the process of condensing steam with the sprays causes a very turbulent effect and high mixing, and you also have your fan coolers that will also permit mixing. This is in addition to the hydrogen mixing system.

Now, what I am about to get into is the burning of all the hydrogen released under certain hypothetical assumptions that were in the proposed rule that the staff has proposed. The 75 percent is the proposed metal-water reaction that I've read inferring all that and what results you get from that.

If you don't have complete mixing, there's another part of that. Then you only burn this part, and you will have parts of the hydrogen that won't completely burn, and the result will be less of a peak pressure.

MR. LIPINSKI: You've eliminated the TMI accident type of scenario from your discussion because the high point vents are being installed on the primary system now to relieve the hydrogen if you should have a contained metal-water reaction within the vessel. So you don't have steam released necessarily if the hydrogen is at the high point.

MR. CERMAK: We looked at the location of the vents coming out of the top of the vessel, and they are into a very open area in the containment where you would get what we feel is very good mixing. We have fans that will give you this mixing. In addition, there is no equipment up in that area.

MR. LIPINSKI: NASA has had some experience where hydrogen has accumulated in the top of their buildings, and they have blown the roofs off. It was not perfect mixing.

MR. CERMAK: What was the mechanism that they blew the roof off?

MR. LIPINSKI: Evidently the equipment had been turned off over the weekend. The hydrogen strategized in the top of the building, and somebody energized the equipment on Monday, and the roof went off.

MR. CERMAK: Well, if I went along with that scenario and put all this hydrogen in the top of the containment, there isn't any equipment up in the top of the containment to ignite the hydrogen.

MR. LIPINSKI: It depends on where the level comes down to. Your crane rails are halfway up.

MR. CERMAK: But we're talking -- first of all, let me get back to the fact that we would expect .1 percent metal-water reaction, and we keep escalating this upward on hypothetical scenarios. Then you have to go and look at the flammability limit and the detonability limit.

The detonability limit for hydrogen and air without any steam is generally considered to be around 19 percent. You have to have a 19 percent concentration. In order to have a 19 percent concentration over the whole dome of the containment, that's quite a bit of metal-water reaction as we will see here in the next slides. So I think it's very probable you get to detonability limits, and then you have to have something to ignite the whole thing.

MR. LIPINSKI: You have lighting systems at the top of the containment. You've got to guarantee that all your electrical systems are intact and there's no arcing or sparking because of a loose connection somewhere.

MR. CERMAK: Well, my basic hypothesis -- I was trying to go along that scenario to answer your question. My basic hypothesis is you get pretty good mixing because of the fan coolers and also the hydrogen mixing.

MR. LIPINSKI: Those fan coolers aren't directed to the top of the containment.

MR. PREBULA: John Prebula, Bechtel.

The containment air coolers draw air from elevation 2,068 and discharge to lower areas of the containment.

MR. LIPINSKI: Where is that relative to the crane rail?

MR. PREBULA: The crane rail is at approximately 2,120. The hydrogen mixing fans are located at elevation 2,047. There are four of them in operation during normal operation. These same fans are Class 1A, and they operate post-accident. They blow air into the containment dome, so that by a replacement mechanism for blowing air out and allowing it to come down the outside shell of the containment to get the proper mixing in the dome. If you actuate the containment sprays you have additional mixing.

MR. CERMAK: The last part I wanted to address is if I put the hypothetical case for large amounts of hydrogen in the large dry containment --

(Slide.)

If I took the 75 percent metal-water reaction that was proposed in the rule I saw about a year ago, in a proposed rule, the results showed 12.5 percent hydrogen by volume in the containment. If I took a constant volume and adiabatic burn of this hydrogen, I would have a pressure increase of 60 PSI. If I did this in a completely dry containment without any steam, that would just equal my design pressure of my containment which is 75 psia. So that for the 75 percent metal-water reaction I am equal to the containment design pressure.

If I went and superimposed a loss of coolant accident on this particular 75 percent metal-water reaction case --

(Slide.)

I am assuming now, and this is very conservative because I'm really below the flammability limit when I do this, I am assuming that the hydrogen burn occurs at the time of the peak containment pressure from the LOCA, which is 47.3 psig at 134 seconds. There is enough steam here that I will not get flammability of the hydrogen. But I made this assumption just to show some degree of conservatism. I then burn the hydrogen and add it to my LOCA peak and I am at 125 psia.

The ultimate static pressure failure for the containment has been estimated to be 175 psia. This follows roughly, and Bechtel's engineering judgment supports this; but if you just take the two and a half multiplier times the design of the containment, which is one 75 psia, you get 187 psia, and 175 is lower than the 187.

Actual analyses have shown to date that pressures above 150 psia, and it can actually be supported today, but the estimate is if the detailed analysis were done, 175 psia could be supported.

MR. ZUDANS: What --

MR. CERMAK: This is about 50 psi below the 125 psia -- sorry -- is about 50 below.

MR. ZUDANS: What is the failure mode associated with 175 psia?

MR. CERMAK: We have Ken Lee of Bechtel here to address the failure mode.

MR. LEE: Kenneth Lee from Bechtel.

The failure mode is in the cylinder of the containment slightly below the transition where the dome meets the cylinder.

MR. ZUDANS: What is the failure mode?

MR. LEE: That's by the rebar reinforcement yielding.

MR. ZUDANS: What about your equipment access hatch and other accesses? Are they able to take all of those pressures as well?

MR. LEE: We have not looked at the penetrations and the personnel lock and equipment hatch at this time. But based on the past few projects that we've looked at, the penetrations do have the capacity higher than the containment shell.

MR. ZUDANS: Of course, your concrete at that stage would be cracking, but you assume that the liner won't.

MR. LEE: That's correct. The liner will be within the strain where it would rupture, well below.

MR. CARBON: John.

MR. ARNOLD: I'm not familiar with the engineering of a containment, but in a steel pressure vessel there is an operating pressure, the pressure at which the vessel is intended to operate, and then above that there is generally a design pressure that allows for the operation of the relief valve safely above the operating pressure.

What pressure is -- when you call it a design pressure, what is that?

MR. LEE: The design pressure is the pressure that we will exceed to calculate above the maximum pressure inside containment. It's the margin we put in exceeding the design pressure.

As Jim mentioned before, we had 47 PSI to calculate a maximum pressure inside containment, and the design pressure for containment is 60 PSI. Then we have an additional factor of 1.2 to assure that we have a margin on top of the design pressure of 60. That is how Jim arrived at the 75 psig.

MR. ARNOLD: Okay.

MR. LEE: That pressure assumes that you have the low loading combinations of earthquake pressure, jet impingement. That is why we have substantial margin beyond that design pressure.

MR. MARK: Is it that the design pressure is approximately at the top of the elastic limit of the members?

MR. LEE: At the specified yield strength of the reinforcement, so you have substantial margin beyond the point where the reinforcement will yield.

MR. ARNOLD: When you take all of that into consideration, what does that mean with respect to the 175 that you had talked about?

MR. LEE: We have not yet made that determination. The calculation that we have made is strictly based on the elastic behavior of the containment. That is why we carry it to the elastic yield limit of the reinforcement. Now, beyond that, the calculations will require a linear analysis, and that is the extra margin that we have not yet identified.

MR. CERMACK: And just to clarify, that is the difference between the 150 that you are now at and the estimated number from the engineering judgment of 175; is that correct?

MR. ZUDANS: Just one more quetstion, since we have a structural person here. Looking at the plant, I could not figure out how your tendons go through the hemispherical part of the head. Do they wrap around and go down to the gallery?

MR. LEE: That is correct.

MR. ZUDANS: Okay. Thank you.

MR. CARBON: Are there questions?

(No response.)

MR. CERMACK: Go ahead.

MR. PASSWATER: I have an answer to one of the questions we did not have an answer to before. The accuracy of the hydrogen monitoring system is .5 percent.

MR. ZUDANS: What is the time delay between sampling and getting to the number? A couple of seconds? You have to pump it through the line.

MR. PASSWATER: It is a very short time. We did not look up the actual figure. It is just a transport time from the atmosphere into the monitor. He is looking for it right now.

The next topic we wanted to present is the capability and reliability of the decay heat removal systems. Hank Schwowerer, who is the technical director for SNUPPS, will speak on that topic.

MR. SCHWOERER: I will just put this up here.

(Slide)

At [illegible] PSAR stage the design basis for safe shutdown [illegible] UPPS plant was extended, hot standby. Using hot s[illegible] in the term in which it is defined in the technical specification, that is you are subcritical but you are at or close to operating temperature.

Prior to Three Mile Island we on our own initiative began to reexamine that design basis. We decided to change that design basis to one of having the capability in the plant to go to cold shutdown with safety-related systems. We really did it for two reasons. One was that we anticipated that cold shutdown was going to be a requirement at the operating license stage. And also we felt that this would enhance the safety of the plant.

The functions that are necessary to go to cold shutdown are listed here in summary fashion. I would just like to tell you rather quickly how these functions are performed with safety-related systems. Heat removal at hot standby is accomplished by natural circulation. When we talkabout cold shutdown with safety-related systems, we are also assuming we have a loss of off-site power.

Heat removal at hot standby then is accomplished by natural circulation within the reactor coolant system of feedwater being supplied to the auxiliary gen[illegible]ator by the steam supply system, steam being relieved from the steam generators by the atmospheric relief valves, which in SNUPPS are Class IE fully qualified valves.

The reactor coolant pump seals under this mode of operation would be cooled by pump cooling water system. And we have made some modifications to the component cooling water system to increase its reliability. Specifically, the pump cooling water system isolates on one of the containment isolation signals.

And we have added some redundant valves in parallel to the existing isolation valves, so that if we were to get an inadvertent isolation of the component cooling water system, we are sure that we can reopen the component cooling water paths.

The next thing you have to do is borate the reactor coolant system. If you are at hot standby, you have to do this within about 24 hours. The way we would do this using safety-related equipment is we would utilize the charging pumps which are capable of pumping against the reactor design pressure. If the boric acid tanks and boric acid transfer pumps were available, we would use that as a source of boration.

As you saw yesterday, though, those tanks and pumps are in the basement of the auxiliary building. They could conceivably be flooded by an earthquake event that might rupture fire protection piping, or they could conceivably be damanged by some nonseismic component that could fall down on them.

So after examining what we might have to do to backfit those tanks and pumps to be fully safety-grade, we decided instead that what we would do would be to use the refueling water storage tank as a source of boration. This means ow that if we are going to borate with the refueling water storage tank, which has water at about 1000 ppm, that we have got to have a capability to let down water so that we can inject into the reactor coolant system this 2000 ppm water from the refueling storage tank.

One of the things then that we have added as part of the cold shutdown package is a fully safety-related letdown system. We take off from the excess letdown line. That is part of the standard Westinghhouse system. We have added some safety-related valves that allow us to drain to the pressurizer relief tank in the containment.

Now, this tank has limited capacity for water and we do not want to rupture the tank. So we have also added some safety-related valves which allow us to drain down from the pressurizer relief tank to the containment itself. So ultimately, the scheme here is that for boration we would plan to inject water from the refueling water storage tank using the storage tank, letting water down to the containment floor.

The next step is the cooldown of the reactor coolant system to -- we want to get it down to about 400 degrees, 350-400 degrees, where we can start operating the RHR system. The method of cooldown is exactly the same method we have used for heat removal at hot standby. We can control the pressurizer -- we can control the atmospheric relief valves on the steam generators to cooldown secondary side and also to cool down the primary side.

Depressurization of the reactor coolant system is the next thing that we have to do. In the procedure that we contemplate here we would want to retain water level in the pressurizer. If we are retaining water level in the pressurizer, we may have hot water retained in the pressurizer, which would tend to hold the pressure up.

We looked at various ways of doing this and finally decided that we would do is change out the power-operated pressure relief valves, which in the standard design were not fully safety-related and not fully qualified.

In the SNUPPS design we have replaced those valves with new valves, electrically operated, that is DC solenoid operated, Class 1E power, fully qualified valves.

Finally, this gets us down to the temperature and pressure at which we can put in the RHR system.

There are two other points here to mention. One concern is if you are going into RHR operation with only safety-related equipment, that you might discharge water from the accumulators, the large safety accumulators, and vent nitrogen into the reactor coolant system. We do not think this would be a problem, but in order to preclude this problem, we have provided safety-related redundant valves that would allow us to vent nitrogen from these accumulators and prevent the injection of the water in these accumulators.

The other potential concern here is that in order to get the RHR system into operation, there ae two valves in series. We have two trains. Each train has two valves in series. The arrangement of the power to those valves is such that we have one train of power powering one valve in each RHR system. This is done to be sure that you always close those valves.

In this kind of an operation, if we postulate that in addition to all the other things that have happened, we should lose one train of power, we would not have the capability to open those valves to the RHR system. The plan for that remote possibility is we would be able to jump our power from one of the electrical penetration rooms around to the other electrical penetration room so that we could be sure that we could open at least one train of our RHR and continue the cooldown process.

Okay. All of these changes, all of these features are described in the FSAR. We can perform all of the actions that I have described from the control room.

I would just like to go on and say a little bit more. We feel that in addition to coming close to meeting the requirements of Regulatory Guide 1.139, which was the one that we anticipated as possibly being a requirement for SNUPPS, we feel this gives us considerable diversity and ability to cope with events beyond design basis events.

One of these, just for example, is the case that has been analyzed by the Westinghouse owners group, which is the loss of all feedwater followed by a loss of all auxiliary feedwater. So it is postulated that you have totally lost the capability to supply feedwater to the steam generators.

This particular scenario has been evaluated by the Westinghouse owners group for a four-loop reactor such as the SNUPPS reactor. It has been determined that for the first 20 minutes nothing particular happens. You boil away water from the steam generators and then at the time that you start to approach dryout of the steam generators, one would then -- there is a standard operating procedure that has been developed that says as you start to approach dryout of the steam generators you initiate feed-and-bleed cooling approach whereby you use the charging pumps to inject water into the reactor coolant system, you use the pressurizer PORVs to let down fluid from the pressurizer.

And for this node of operation it is possible to keep the core covered and avoid any core damage. We feel that with the safety-related PORVs we have real capability to do this under almost any situation.

MR. ZUDANS: Are your PORVs big enough to remove all the residual heat?

MR. SCHWOERER: Yes, they are.

MR. ZUDANS: How manydo you have?

MR. SCHWOERER: We have two.

MR. ZUDANS: Where does the discharge go?

MR. SCHWOERER: The discharge also goes to the pressurizer relief tank. Once again, the pressurizer relief tank is not large enough to contain all the steam, so what would happen in all this mode is we would be transfering water from the reactor coolant system to the containment.

MR. ZUDANS: Right.

MR. SCHWOERER: Now, one of the things I failed to mention is in the -- well, okay.

MR. ZUDANS: But how long could you do that? You cannot keep on transferring to the containment. What about the feed capacity?

MR. SCHWOERER: Well, we have enough -- as part of these modifications that we made, we made certain that we did not have any safety-related equipment in the containment below a certain level. I am not sure just offhand what the number is. Is it 600,000 gallons of water?

MR. STRIGHT: Yes.

MR. SCHWOERER: We are capable of taking 6000 gallons of water into the containment without flooding any safety-related equipment. The capacity of the reactor coolant system is approximately 100,000 gallons.

MR. ZUDANS: But to feed that many gallons, where do you get the water from?

MR. SCHWOERER: The refueling water storage tank.

MR. ZUDANS: How big is that?

MR. SCHWOERER: Well, I am not just sure.

MR. PREBULA: The short capacity is 379,000 gallons.

MR. ZUDANS: So you already use some of it for your boration.

MR. SCHWOERER: I think you get into -- I give you the same numbers that Steve Miltenberger showed yesterday, because again now we are coping with decay heat. His numbers showed that we have many, many hours to -- we could carry this one for many, many hours.

We are talking about a very hypothetical situation. You know, we are talking about a case where we have lost off-site power, we have lost all of our feedwater and our auxiliary feedwater. We have gone pretty far down a road that is very improbable. I just cite this to give you an example of some of the diversity that exists in the design.

MR. ZUDANS: So you have several hours to recover your off-site power.

MR. SCHWOERER: Many hours, yes.

MR. ZUDANS: Your diesels are running, though.

MR. SCHWOERER: We assume our diesels are running, yes.

MR. CARBON: I was mixed up on that. When you talked about item number 1, I thought you were talking about natural circulation. I lost out on what happened.

MR. SCHWOERER: Yes, sir, I am. The reactor coolant pumps are not powered by the diesels. Reactor coolant pumps are powered by off-site.

MR. ZUDANS: They are much too big, and you do not have extra spray for the pressurizer either.

MR. SCHWOERER: Well, in this thing we are looking at, fully safety-related systems, one of the difficulties is the chemical volume control system, which would normally be supplying water to the reactor coolant system, this has many air-operated valves that is supplied by the instrument air system, which is not a fully safety-related system.

So we assume that all the air-operated valves failed and the valves failed in open or closed conditions depending upon the particular design. But we do not have auxiliary spray available to the pressurizer, because that is supplied by this containment supply system through a valve that would fail closed.

MR. ZUDANS: Now, are there any of the valves in any of the safety-related systems, such as the feedwater system or RHR or any other one that is controlled by instrument air as a driving power to open or close the valves in any of the safety-related systems? Are there any valves whose operation depends on instrument air?

MR. SCHWOERER: No.

MR. ZUDANS: That is why you kicked out the chemical control?

MR. SCHWOERER: That's right.

MR ZUDANS: The cooldown on item 3, cooldown of the RCS, you do that by controlling the PORVs on the pressurizer; right?

MR. SCHWOERER: No. You do that by controlling the PORVs on the steam generator, what we call the atmospheric relief valves.

MR. ZUDANS: I am sorry, I really wanted to talk about item 4. This is where you replace your regular PORVs with Class 1E PORVs and those you can open and close and they will survive any environment; is that right?

MR. SCHWOERER: Yes.

MR. ZUDANS: Is that conclusion based on some test valves, or are these valves to be designed yet?

MR. SCHWOERER: They are new valves. These are valves manufactured by Air Research. I do not believe the entire test program has been completed. They have been tested as part of the EPRI safety relief valve program. The functional test has been performed under that program and shows that they perform as designed.

There is also a seismic and environmental qualification program that is being performed by Westinghouse, and I am just not aware of whether all of those tests have been completed yet.

MR. ZUDANS: Would they include testing against discharging just all water, or a mixture of water, or just all steam, or all the combinations?

MR. SCHWOERER: I think I need a little bit of help on this. Jim, maybe you know, on the EPRI valve test program, I do not know what the test conditions were for these particular valves.

MR. CERMACK: First of all, the steam testing was done, the steam testing was done at the Marshall facility of the Duke Power Company. The water testing of the valves at the Wylie Laboratories in California. And the water-followed-by-steam testing was done also at the Wylie

Laboratories.

In all cases the test results were very good, excellent, virtually no leakage of the valves.

The environmental qualification of the valve, as Frank was saying, is being conducted by Westinghouse at the Air Research facility.

MR. SCHWOERER: My speculation would be that that is probably not completed as yet. The valves were scheduled to be delivered to the Callaway plant in December of this year.

MR. CERMACK: That is correct. The qualification tests will be then.

MR. ZUDANS: I am going to speculate now. If you also lost the diesel, you could still stay in a hot shutdown state but you could not go to cold, naturally.

MR. SCHWOERER: Well, if you lose a diesel, if you lose off-site power and you lose the diesel, that was the case discussed by Steve Miltenberger yesterday, the concern is that you do not have any way of making up water to the reactor coolant system.

So then the concern is that you must minimize leakage from that system. One of the concerns is what happens to the reactor coolant pump seals. That was discussed in that context. We are not assuming that we lose the diesel in this case. We are assuming that we have the diesel. You need the diesel to run the charging pumps.

Actually, the first place we need the diesel is here (indicating). When we decide we are going to borate and we need to have some ability to charge borated water into the reactor coolant pump system.

Also, part of our thought here is that under normal procedures when you are taking a plant from a hot condition down to a cold condition, you borate to the cold shutdown concentration before you start cooling down. We wanted to have that capability.

It is possible, of course, to borate while you are cooling down, and this would still be safe. One of the advantages of doing that is as you cool down you make volume to accept the boric acid. With the refueling water storage tank as the source of borated water, however, you cannot get enough boron in if you are just taking advantage of shrinkage of the whole system.

The other thing is we wanted to maintain pressurizer level throughout this entire evolution.

MR. ZUDANS: Thank you.

MR. CARBON: Are there any other questions?

(No response.)

MR. CARBON: We thank you.

Does the Staff have any final comments of any kind to make?

MR. EDISON: No, sir.

MR. CARBON: Anything else?

MR. PASSWATER: No, sir. That concludes our presentation.

MR. CARBON: I guess I would ask the committee members is there any question on bringing this to the full committee next week? Carson?

MR. MARK: I see no reason why not.

MR. CARBON: Fine. We will expect you to come to the full committee meeting next week, Thursday afternoon.

In terms of the topics, have you put notes or anyting together?

MR. ARNOLD: Do we have time to have a little recess to go over them? I think we might want to look at them first, if that is a reasonable suggestion.

MR. CARBON: You mean for us to discuss among ourselves or to break and give a little thought to it?

MR. MARK: We could really end this meeting, I think, and confer with the Staff and the Applicant afterwards about the topics at the next meeting on this one.

MR. CARBON: Let us take that approach then of adjourning the meeting, and the subcommittee will continue to discuss the topics and the timing that we would like to ask you to present next week.

Before we do adjourn the meeting, I would like to make one statement to you, Mr. Schnell. It has been my impression, rightly or wrongly, that you and your people have sort of bent over to try and be open and candid and honest and aboveboard and not mislead us. I cannot really say that about all of the groups that have met before us in the past. And I simply wanted to recognize this. That is my impression, and I commend you fo it. I just think it is the best way to go at it.

MR. SCHNELL: Dr. Carbon, thank you very much for those remarks. We continue this very important step in the licensing process of obtaining our operating permit for Callaway. And we certainly want to be responsive to all of your questions, and we appreciate that remark.

MR. CARBON: Well, we will adjourn the meeting then. The subcommittee will continue to come up with the topics for presentation next week.

(Brief recess.)

MR. CARBON: What we would like to ask, if you are set here, is the following kind of things, which I do not have into a really polished list, but I think you can make out from it. We would ask for about a total of 30 minutes of organization and management by UE, by the Applicant, organization, technical competence, experience levels, with some technical discussion on SNUPPS.

We would ask that you hold down the organizational aspect of SNUPPS, but we think tha'. the technical backup flavor ought to come through. And we would urge that.

In connection with experience levels, we wonder if you could put some sort of quick chart together which the committee could see. We are going to have lots of questions on the experience level.

If you could, show the people in the organization both at the management engineering level as well as in the operations level. And perhaps indicate their positions and their years or professional experience and the years of nuclear experience and the years of commercial nuclear experience, one column or something showing where your counting Callaway and where you are not counting Callaway so that we can look at this and say, "Gee, someone has five years of professional experience. This was at Callaway or it was three at Callaway and two at some place else."

We do not really care too much about the names of the people except perhaps the management people. And if they have ten engineers, you do not have to name them all, or five reactor supervisors, do not name them. But indicate their positions and the numbers of people and so on. We will have considerable questions in that area.

Another aspect there that we will be concerned about has to do with the startup. We wonder somewhat in more detail than you explained before what the relationship during startup between the startup supervisor and the plant superintendent will be. We are not real clear on this. We are wondering who will have what specific responsibility and authority and so on.

Then in addition, your startup crew, as you indicated, will have the 44 total professional people, a dozen or so of these being your people. And we would welcome additional information than we feel we have as to how these people mesh together, how your people are going to get the experience that you want them to get, and how is going to call the shots during the startup, what authority do the non-UE people have, those being the ones with the msot experience as contrasted to your own people.

Is that clear, what I am saying there?

MR. PASSWATER: Yes.

MR. CARBON: Going on then to the second topic, and again these are not in order, perhaps 20 minutes or so of operations staffing and training, current status of staffing, selection and training of operators, something on your simulator, 15 minutes on emergency planning, coordination with FEMA and the state, emergency support facilities, population projections. We have that down here, but population projections, 10 seconds isabout adequate.

Would you try and combine the topic, the instrumentation to follow the course of an accident, Jim Cermack's discussion, with the emergency opeating procedures discussion by Mr. Neuhalfen, combine those into one single presentation of not more than 45 minutes.

We would like certainly the discussion of open items both by the Staff and UE, and perhaps we could squeeze those into about 45 minutes, because they are not really very earthshaking and I do not think we need too much discussion.

Perhaps a 15-minute discussion on the control room and perhaps about 15 minutes of discussion on the last topic, the decay heat removal by Mr. Schwoerer, that topic anyway.

Now, this does not come out to quite four hours. We want to limit the total time at the full committee meeting to four hours. And Mr. Majors will be in touch with you to expand some of these times and juggle them briefly.

MR. PASSWATER: Dr. Carbon, are these our presentation times, or does this include time for questions?

MR. CARBON: It is going to have to include the time for questions. I think if you add up what I have cited here, it probably comes out to about three hours and 20 minutes if my arithmetic was correct. And then I will need 15 minutes or so. So that comes out nominally to 3-1/2 hours already, and we will expand this a little to take up the total of four. But that has to include everything.

Let me ask if you would and if you would, when you show your organization chart, it seems like at so many of our meetings so much time is spent telling what person holds what position. If you could show an organization chart with manager of engineering or manager of something, simply put their name on there and then you will not have to tell us who it is. We can read it and save a little bit of time.

Have I left anything out?

MR. MARK: You did not mention the situation involved in the station blackout, and yet there is very likely to be a question on that. We would hope you would have someone who would be able to describe the situation even if it is not scheduled.

MR. CARBON: I have left three or four topics off. I will report to the committee on those topics, such as AC/DC power reliability, hydrogen control, your systems interaction discussion. They are not on the agenda here. But do have people prepared to answer questions in case the committee members do have such questions -- and they will, there will be questions that are not covered here.

What other things are there?

MR. PASSWATER: One thing I understand that has been on many recent committee meetings has been the item of security, plant security.

MR. CARBON: I guess probably we will want some discussion on that. We have your security plan, and I plan to read it next week, but I have not done so yet. Let us schedule --

MR. EDISON: I might interject, there is no issue on this as far as the Staff is concerned.

MR. RAY: I think we should give them 10 or 15 minutes.

MR. MARK: It is not the question of what we want. The fact that it is mentioned that the thing has been looked and it seems to be well within the pattern of other provisions for security. If we go much further than that, we are going to close the lousy meeting and that takes 10 minutes to get in and out.

MR. CARBON: Let me suggest this. We will not put it on the agenda, but have an individual there who could answer questions if we get into them. And if we want to close the meeting, we can always do it, but it is unlikely that we will.

MR. CAPONE: I know it is hard to depict, but would that be more along the line of security during operations or the design of the security system itself? We can bring people to go both ways. We can discuss the overall design, how it has been put together, or are you more concerned about how the security --

MR. CARBON: Lots of people ask the question, "How about the card system and the buddy system," but also other questions.

MR. MARK: What is that? Part 73 something or other?

MR. EDISON: Yes.

MR. MARK: I think it is 73. The fact that you have met the requirements of this -- which I presume is the case?

MR. EDISON: Right.

MR. MARK: And that you are prepared to enumerate and keep a tight control of vital areas and use keycards, if that is the right word, if that is what they are doing, and you have all of the standard fences and guard levels and alarms, I think you can say that much in an open meeting.

But if you get down to describing drawings, "Here is where that is, there is where that is," then we have probably got to close the meeting.

MR. PASSWATER: Right. I understand.

MR. MARK: So I think we ought to be able to get through without needing to close it, by giving the general status of things.

Does that sound right?

MR. EDISON: If we do that, it will not be necessary for me to bring the Staff security reviewer to the meeting, because I was present with him and walked down the site and we have that well documented.

MR. MARK: You would be prepared to say what had been looked and that seems to cover all the points that we usually try to cover, if that is the case. To say that is all right. To describe them is not.

MR. EDISON: Right.

MR. CARBON: Are there any other questions?

MR. PASSWATER: I do not think so at this time.

MR. CARBON: Fine. We will see you all next week.

MR. PASSWATER: Thank you.

(Whereupon, at 12:40 p.m., the Subcommittee adjourned.)

* * *

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ACRS/CALLAWAY UNIT 1 PLANT TOUR AND MEETING

Date of Proceeding: November 5, 1981

Docket Number: ____________________

Place of Proceeding: Columbia, Missouri

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

Jane W. Beach

Official Reporter (Typed)

Jane W. Beach

Official Reporter (Signature)

SUMMARY OF

CONFIRMATORY ITEMS

FROM SAFETY EVALUATION

REPORT FOR CALLAWAY

PLANT, UNIT 1

TI 11-5-81

CONFIRMATORY ITEMS FROM SER

CHARACTERISTICS

- STAFF AGREES WITH APPLICANT'S DESIGN
- NEED TO COMPLETE DOCUMENTATION
- STAFF WANTS TO VERIFY IMPLEMENTATION AT SITE
- ANALYSES NEARLY COMPLETE AND FINAL TOUCHES WILL NOT CHANGE RESULT
- INSPECTION OF TEST RESULTS

CONFIRMATORY ITEMS FROM SER

STATUS

- TOTAL OF 27 ITEMS + 12 TMI-RELATED ITEMS
- TWO ITEMS MAY BE CONFIRMED BY 11-12-81
- MOST IN 1982-1983

SUMMARY OF

LICENSE CONDITIONS

FROM SAFETY EVALUATION

REPORT FOR

CALLAWAY PLANT, UNIT 1

LICENSE CONDITIONS FROM SER

CHARACTERISTICS

- ACTIONS THE APPLICANT MUST TAKE (OR CONDITIONS TO BE MAINTAINED) OVER THE OPERATING LIFE OF THE PLANT
- TECHNICAL SPECIFICATIONS

LICENSE CONDITIONS FROM SER

STATUS

- 20 LICENSE CONDITIONS
- EXPECT MOST TO BE IMPLEMENTED PRIOR TO LICENSING AND THEREFORE WILL NOT BECOME LICENSE CONDITIONS

end T1

OPEN ITEMS FROM SER

&

SUMMARY OF SER CONFIRMATORY ISSUES

&

SUMMARY OF SER LICENSING CONDITIONS

R. L. Stright, Licensing Manager - SNUPPS

11-5-81

Tape
1

PREPARATION OF EMERGENCY OPERATING PROCEDURES

A. P. Neuhalfen, Superintendent of Operations

1-A 11-5-81

PREPARATION OF EMERGENCY OPERATING PROCEDURES

I. Current Development

II. Format Philosophy

III. Example Format

IV. Procedure Structure

V. Guideline Lists

VI. Status Trees

VII. Co-ordinated Use

VIII. E-0 Procedures

IX. Summary

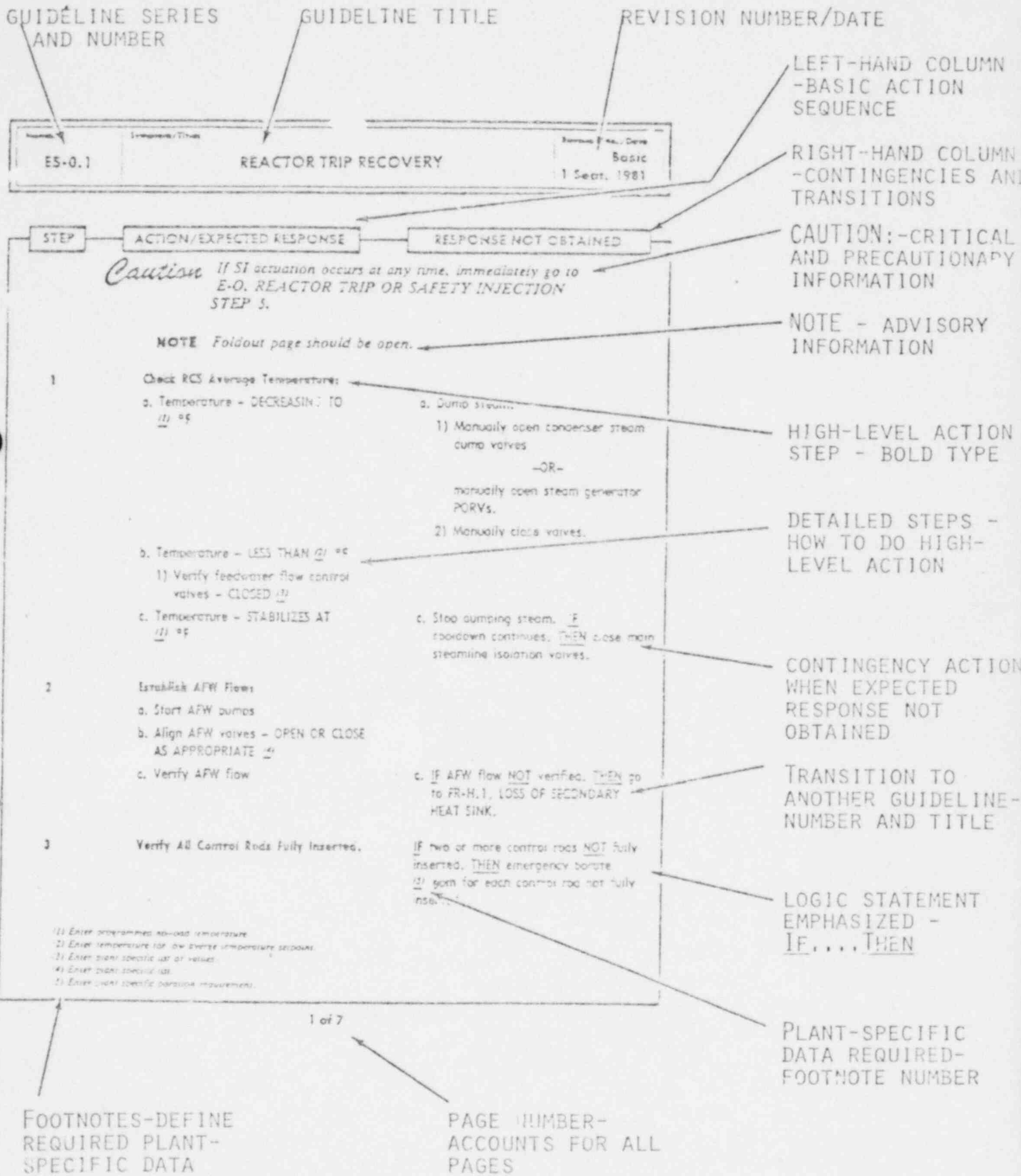


FIGURE 1. FORMAT EXAMPLE

PROCEDURE STRUCTURE

I. Optimal Recovery

1. Emergency Procedures (E)
2. Emergency Subprocedures (ES)
3. Emergency Contingency Actions (ECA)

II. Functional Recovery

1. Critical Safety Function Status Trees
2. Functional Restoration Guidelines (FRG)

TABLE 1A

OPTIMAL RECOVERY GUIDELINES

E-0 REACTOR TRIP OR SAFETY INJECTION

- ES-0.1 REACTOR TRIP RECOVERY
- ES-0.2 NATURAL CIRCULATION COOLDOWN
- ES-0.3 SI TERMINATION FOLLOWING SPURIOUS SAFETY INJECTION

E-1 LOSS OF REACTOR COOLANT

- ES-1.1 SI TERMINATION FOLLOWING LOSS OF REACTOR COOLANT
- ES-1.2 POST-LOCA COOLDOWN AND DEPRESSURIZATION
- ES-1.3 TRANSFER TO COLD LEG RECIRCULATION FOLLOWING LOSS OF REACTOR COOLANT
- ES-1.4 TRANSFER TO HOT LEG RECIRCULTATION

E-2 LOSS OF SECONDARY COOLANT

- ES-2.1 SI TERMINATION FOLLOWING LOSS OF SECONDARY COOLANT
- ES-2.2 TRANSFER TO COLD LEG RECIRCULTATION FOLLOWING LOSS OF SECONDARY COOLANT

E-3 STEAM GENERATOR TUBE RUPTURE

- ES-3.1 SI TERMINATION FOLLOWING STEAM GENERATOR TUBE RUPTURE
- ES-3.2 ALTERNATE SGTR COOLDOWN
- ES-3.3 SGTR WITH SECONDARY DEPRESSURIZATION

TABLE 1B

EMERGENCY CONTINGENCY ACTIONS

ECA-1 ANTICIPATED TRANSIENTS WITHOUT SCRAM

ECA-2 LOSS OF ALL AC POWER

- ECA-2.1 LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED
- ECA-2.2 LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED

ECA-3 SGTR CONTINGENCIES

TABLE 2

FUNCTION RESTORATION GUIDELINE SET (FRGs)

FR-S.1	RESPONSE TO NUCLEAR POWER GENERATION
FR-S.2	RESPONSE TO LOSS OF CORE SHUTDOWN
FR-P.1	RESPONSE TO RCS OVERPRESSURIZATION
FR-P.2	RESPONSE TO HIGH RCS PRESSURE
FR-C.1	RESPONSE TO INADEQUATE CORE COOLING
FR-C.2	RESPONSE TO POTENTIAL LOSS OF CORE COOLING
FR-C.3	RESPONSE TO SATURATED CORE COOLING CONDITIONS
FR-I.1	RESPONSE TO PRESSURIZER FLOODING
FR-I.2	RESPONSE TO LOW SYSTEM INVENTORY
FR-I.3	RESPONSE TO VOIDS IN REACTOR VESSEL
FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK
FR-H.2	RESPONSE TO LOW STEAM GENERATOR LEVEL
FR-H.3	RESPONSE TO LOSS OF NORAML STEAM DUMP CAPABILITY
FR-Z.1	RESPONSE TO CONTAINMENT ABOVE DESIGN PRESSURE
FR-Z.2	RESPONSE TO HIGH CONTAINMENT PRESSURE
FR-Z.3	RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL
FR-Z.4	RESPONSE TO HIGH HYDROGEN CONCENTRATION IN CONTAINMENT
FR-Z.5	RESPONSE TO CONTAINMENT FLOODING

(1) SUBCRITICALITY (S-SERIES)

(2) RCS INTEGRITY (P-SERIES)

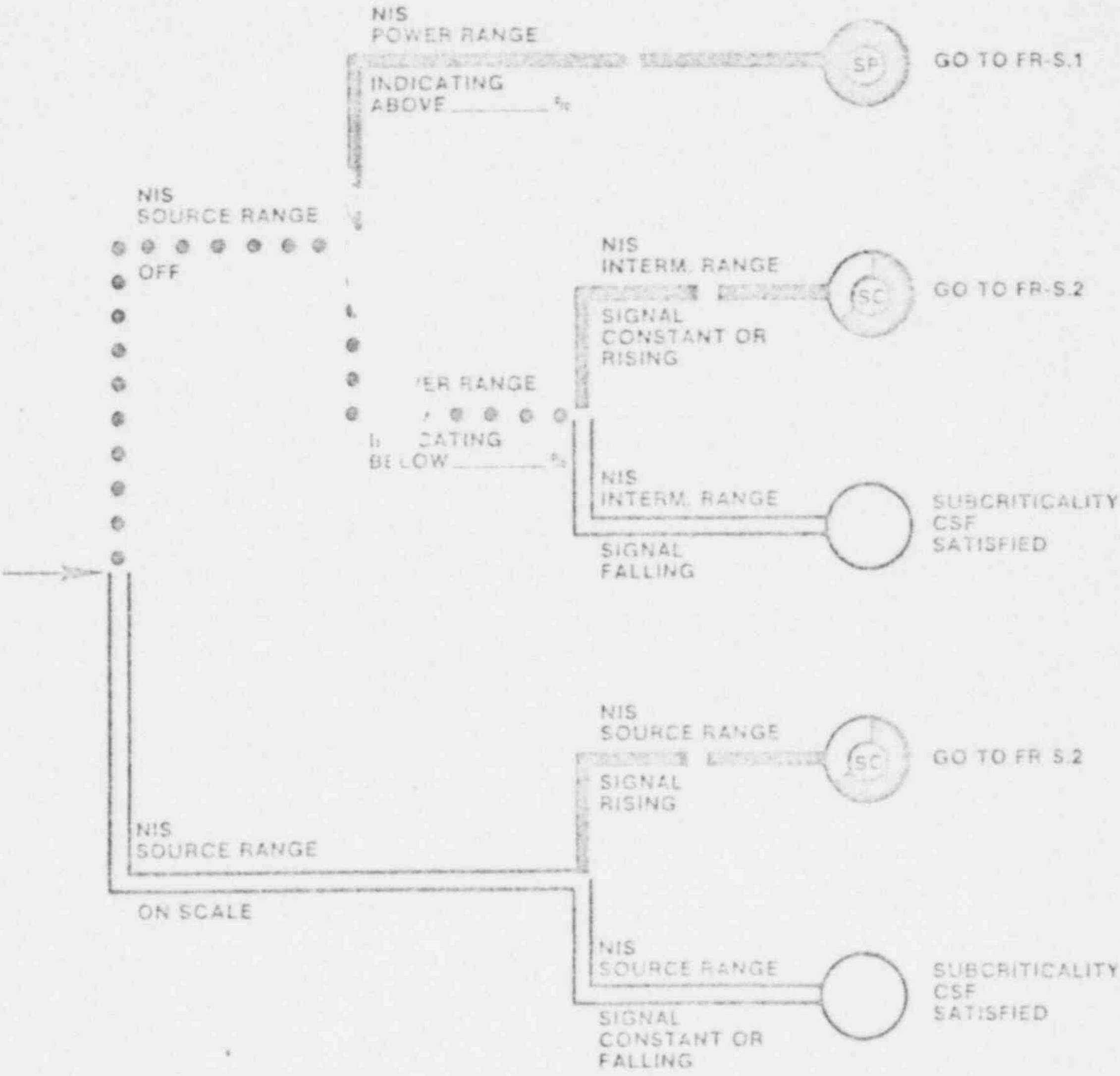
(3) CORE COOLING (C-SERIES)

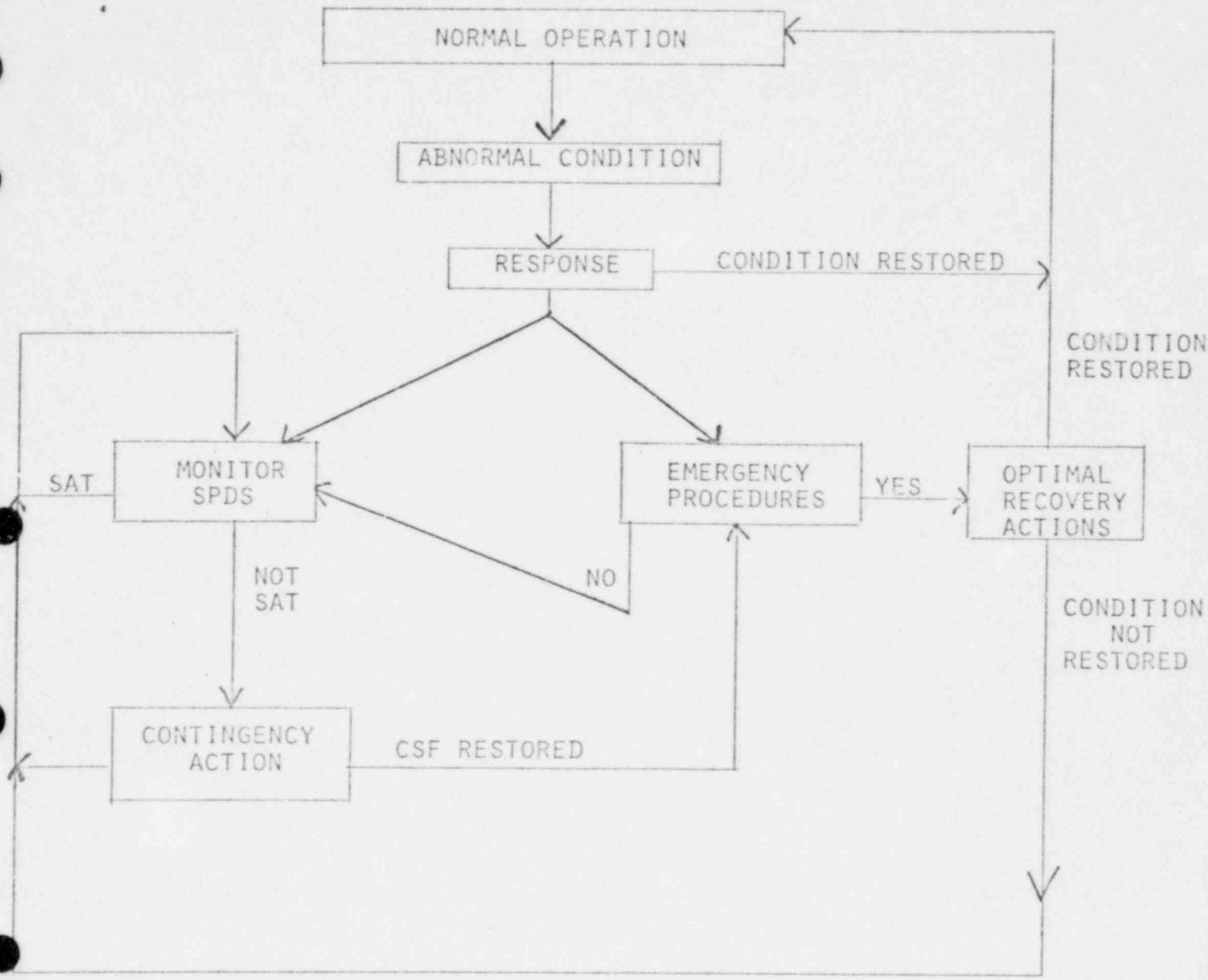
(4) RCS INVENTORY (I-SERIES)

(5) CORE HEAT SINK (H-SERIES)

(6) CONTAINMENT INTEGRITY (Z-SERIES)

SUBCRITICALITY




NORMAL OPERATION
ABNORMAL CONDITION
RESPONSE
CONDITION RESTORED
CONDITION RESTORED
MONITOR SPDS
SAT
EMERGENCY PROCEDURES
YES
OPTIMAL RECOVERY ACTIONS
NOT SAT
NO
CONDITION NOT RESTORED
CONTINGENCY ACTION
CSF RESTORED

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

EMERGENCY INSTRUCTION E-O
REACTOR TRIP OR SAFETY INJECTION

A. PURPOSE

The purpose of this guideline is to verify proper response of the automatic protection systems following actuation of a REACTOR TRIP or SAFETY INJECTION; and to assess plant conditions and identify the appropriate recovery guideline.

B. SYMPTOMS: (1)

I. Following are symptoms of a reactor trip:

a. Any reactor trip annunciator lit
b. Rapid decrease in neutron level indicated by nuclear instrumentation
c. All shutdown and control rods are fully inserted. Rod bottom lights are lit
d. Rapid decrease in unit load to zero power

II. Following are symptoms of reactor trip *and* safety injection:

a. Any SI annunciator lit
b. SI pumps in service
c. [Enter other plant specific symptoms]

(1) Plant should modify this typical list to be consistent with plant features.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE • *Circled numbers show IMMEDIATE ACTION steps.* • *Foldout page should be open.*	
1	**Verify Reactor Trip:** • All rod bottom lights – LIT • All rod position indicators – ZERO • Neutron flux – DECREASING	a. Manually trip reactor. IF reactor will NOT trip, THEN go to ECA-1, ANTICIPATED TRANSIENT WITHOUT SCRAM.
2	**Verify Turbine Trip:** • All turbine stop valves – CLOSED	• Manually trip turbine.
3	**Verify AC Emergency Busses Energized:** • AC emergency bus voltage – NORMAL	IF NOT energized, THEN go to ECA-2, LOSS OF ALL AC POWER, STEP 3.
4	**Check If SI Is Actuated:** a. [Enter plant specific means]	IF NOT actuated, THEN go to ES-0.1, REACTOR TRIP RECOVERY.
5	**Verify Feedwater Isolation:**	
	a. Flow control valves – CLOSED	a. Manually close valves.
	b. Flow control bypass valves – CLOSED	b. Manually close valves.
	c. Feedwater isolation valves – CLOSED	c. Manually close valves.
	d. Steam generator blowdown isolation valves – CLOSED	d. Manually close valves.
6	**Verify Containment Isolation Phase A:** a. Isolation phase A valves – CLOSED (1)	a. Manually close valves.
7	**Verify AFW Pumps Running:**	
	a. Motor-driven pump breaker indicator lights – LIT	a. Manually start pumps.
	b. Turbine-driven pump steam supply valves – OPEN	b. Manually open valves.

(1) Enter plant specific list.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	**Verify AFW Valve Alignment:**	
	a. AFW valves - PROPER EMERGENCY ALIGNMENT (1)	a. Manually open or close valves as appropriate.
9	**Verify SI Pumps Running:**	
	a. Charging/SI pump breaker indicator lights - LIT	a. Manually start pumps.
	b. High-head SI pump breaker indicator lights - LIT	b. Manually start pumps.
	c. Low-head SI pump breaker indicator lights - LIT	c. Manually start pumps.
10	**Verify SI Valve Alignment:**	
	a. SI valves - PROPER EMERGENCY ALIGNMENT (1)	a. Manually open or close valves as appropriate.
11	**Verify CCW Pumps Running:**	
	a. CCW pump breaker indicator lights - LIT	a. Manually start pumps.
12	**Verify Service Water Pumps Running:**	
	a. Service water pump breaker indicator lights - LIT	a. Manually start pumps.
13	**Verify Containment Ventilation Isolation:**	
	a. Damper indicator lights - CLOSED	a. Manually close damper.

[Appropriate steps for verification of other essential equipment as required by the specific plant design should be placed after step 13.].

(1) Enter plant specific list.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Caution *If SI flow cannot be verified, symptoms should be monitored for FR-C.1, INADEQUATE CORE COOLING.*

(14)	**Verify SI Flow:**	
	a. Charging/SI pump flow indicator – CHECK FOR FLOW	a. Manually start pumps and align valves as appropriate.
	b. IF RCS pressure is less than (1) psig, THEN check high-head SI pump flow indicators – CHECK FOR FLOW	b. Manually start pumps and align valves as appropriate.
	c. IF RCS pressure is less than (2) psig, THEN check low-head SI flow indicators – CHECK FOR FLOW	c. Manually start pumps and align valves as appropriate.

Caution *Do not throttle AFW flow until the water level is above the top of the U-tubes.*

(15)	**Verify AFW Flow:**	
	a. AFW flow indicators – CHECK FOR FLOW	a. IF AFW flow NOT verified, THEN go to FR-H.1, LOSS OF SECONDARY HEAT SINK.
(16)	**Verify RCS Heat Removal:**	
	a. RCS average temperature – DECREASING TO (3) °F	a. Dump steam: 1) Manually open condenser steam dump valves. —OR— 2) Manually open steam generator PORVs.

(1) Enter plant specific shutoff pressure of high-head SI pumps.
(2) Enter plant specific shutoff pressure of low-head SI pumps.
(3) Enter temperature for programmed no-load temperature.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17	**Check Containment Pressure:**	
	a. Pressure has remained below (1) psig	a. IF pressure has remained below (1) psig, THEN verify main steam isolation valves closed. IF NOT closed, THEN manually close valves.
	b. Pressure has NOT gone ABOVE (2) psig	b. IF pressure has gone above (2) psig, THEN: 1) Verify containment spray initiated. IF NOT initiated, THEN manually initiate. 2) Verify containment isolation phase B initiated. IF NOT initiated, THEN manually initiate. 3) Stop all RCPs.
18	**Check RCS Pressure:**	
	a. Pressure – GREATER THAN (3) PSIG	a. IF less than (3) psig, THEN go to step 27.
	b. Pressure – STABLE or INCREASING	b. IF decreasing, THEN go to step 27.
19	**Check Containment Temperature:**	
	a. Containment temperature – NORMAL	a. IF high, THEN go to step 27.
20	**Check Containment Pressure:**	
	a. Containment pressure – NORMAL	a. IF high, THEN go to step 27.
21	**Check Containment Radiation:**	
	a. Containment radiation – NORMAL	a. IF high, THEN go to step 27.
22	**Check Containment Recirculation Sump Level:**	
	a. Containment recirculation sump level – NORMAL	a. IF high, THEN go to step 27.

(1) Enter plant specific Hi-2 pressure setpoint.
(2) Enter plant specific Hi-3 pressure setpoint.
(3) Enter plant specific low pressure reactor trip setpoint.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23	**Check Steam Generator Blowdown Radiation:**	
	a. Radiation – NORMAL	a. IF high, THEN go to step 27.
24	**Check Condenser Air Ejector Radiation:**	
	a. Radiation – NORMAL	a. IF high, THEN go to step 27.
25	**Check If SI Can Be Terminated:**	
	a. RCS pressure – GREATER THAN 2000 PSIG AND INCREASING	a. DO NOT TERMINATE SI. Go to step 27.
	b. Pressurizer level – GREATER THAN (1) %	b. DO NOT TERMINATE SI. Go to step 27.
	c. RCS subcooling – GREATER THAN (2) °F	c. DO NOT TERMINATE SI. Go to step 27.
	d. Secondary heat sink:	
	1) Total AFW flow to non-faulted steam generators – GREATER THAN (3) GPM	d. IF neither condition is satisfied, THEN DO NOT TERMINATE SI. Go to step 27.
	-OR-	
	2) Wide range level in at least one non-faulted steam generator – – GREATER THAN (4) %	
26	**Terminate SI:**	
	a. Go to ES-0.3, SI TERMINATION FOLLOWING SPURIOUS SI	

(1) Enter plant specific no-load value.

(2) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation tables.

(3) Enter plant specific value derived from background document.

(4) Enter plant specific value which is above top of steam generator U-tubes.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
27	**Check If RCS Depressurization Can Be Stopped:**	
	a. Pressurizer spray valves - CLOSED	a. Manually close valves.
	b. Pressurizer PORVs - CLOSED	b. Manually close valves. <u>IF</u> any valve cannot be closed, <u>THEN</u> manually close its block valve.

Caution *Seal injection flow should be maintained to all RCPs.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
28	**Check If RCPs Should Be Stopped:**	
	a. SI running - CHECK FOR FLOW OR PUMP BREAKER INDICATOR LIGHTS LIT • Charging/SI -OR- • High-head SI	a. DO NOT STOP RCPs. Go to step 29.
	b. RCS pressure - EQUAL TO OR LESS THAN (1) PSIG	b. DO NOT STOP RCPs. Go to step 29.
	c. Stop All RCPs	

(1) Enter plant specific value, derived from background document.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
29	**Check For Secondary Integrity:**	
	a. All steam generator pressures – APPROXIMATELY EQUAL	a. IF pressure 100 psi lower in one steam generator than the others, THEN go to E-2, LOSS OF SECONDARY COOLANT.
	b. All steam generator pressures – GREATER THAN (1) PSIG	b. IF any steam generator pressure less than (1) psig, THEN go to E-2, LOSS OF SECONDARY COOLANT.
30	**Check For RCS Integrity:**	
	a. Containment pressure – NORMAL	a. IF high, THEN go to E-1, LOSS OF REACTOR COOLANT.
	b. Containment radiation – NORMAL	b. IF high, THEN go to E-1, LOSS OF REACTOR COOLANT.
	c. Containment recirculation sump level – NORMAL	c. IF high, THEN go to E-1, LOSS OF REACTOR COOLANT.
31	**Check For RCS To Secondary Integrity:**	
	a. Condenser air ejector radiation – NORMAL	a. IF high, THEN go to E-3, STEAM GENERATOR TUBE RUPTURE.
	b. Steam generator blowdown radiation – NORMAL	b. IF high, THEN go to E-3, STEAM GENERATOR TUBE RUPTURE.

(1) Enter plant specific value corresponding to low steam pressure SI setpoint.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
32	**Continue To Evaluate Plant Conditions:** a. Monitor critical safety function status trees -AND- b. Continue with this guideline	
33	**Check RCS Hot Leg Temperatures:**	
	a. IF at least one RCP is running - HOT LEG TEMPERATURE STABILIZES AT APPROXIMATELY (1) °F	a. IF temperature is decreasing in an uncontrolled manner, THEN close main steamline isolation valves and bypass valves.
	b. IF RCPs are NOT running- HOT LEG TEMPERATURE STABILIZES AT APPROXIMATELY (2) °F	b. IF temperature is decreasing in an uncontrolled manner, THEN close main steamline isolation valves and bypass valves.
34	**Check Steam Generator Levels:**	
	a. Narrow range level - GREATER THAN (3) %	a. IF less than (3) %, THEN maintain full AFW flow until narrow range level is greater than (3) %.
	b. Throttle AFW flow to maintain narrow range level at (4) %.	
35	**Check PRT Conditions-NORMAL.**	IF PRT conditions abnormal, THEN evaluate cause of abnormal conditions.

(1) Enter temperature for programmed no-load temperature.

(2) Enter temperature corresponding to expected hot leg temperature on natural circulation.

(3) Enter plant specific value showing level just in the narrow range including allowances for normal channel accuracy, post-accident transmitter errors and reference leg process errors.

(4) Enter plant specific value corresponding to no-load steam generator level including allowances for post-accident transmitter errors and reference leg process errors.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
36	**Check If Low-Head SI Pumps Should Be Stopped:**	
	a. RCS pressure – GREATER THAN (1) PSIG AND STABLE OR INCREASING	a. IF RCS pressure low or decreasing, THEN return to step 28.
	b. Reset SI	
	c. Stop low-head SI pumps and place in standby	

Caution *If RCS pressure drops below (1) psig, the low-head SI pumps must be manually restarted to supply water to the RCS.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
37	**Rediagnose Plant Conditions:**	
	a. Return to step 18.	

— END —

(1) Enter plant specific shutoff head of low-head SI pumps.

STAFF POSITION

ON HYDROGEN CONTROL

FOR LARGE DRY

CONTAINMENTS:

CALLAWAY PLANT, UNIT 1

TS 12-5-81

HYDROGEN CONTROL

- o STAFF POSITION FOR LARGE DRY CONTAINMENTS
- o CALLAWAY DESIGN

STAFF POSITION ON HYDROGEN
CONTROL FOR LARGE DRY CONTAINMENTS

- 10 CFR 50.44
  (5% METAL-WATER REACTION OR 5 X ECCS CALCULATION)

- CONSISTENT WITH FULL POWER LICENSING OF N. ANNA 2, SALEM 2, FARLEY 2, TMI-1 RESTART

- PROPOSED INTERIM RULE NOT NOW IMPOSED

CALLAWAY DESIGN

- 2.5 MILLION FT^3 CONTAINMENT FREE VOLUME
- TMI-2 TYPE EVENT (30-50% METAL-WATER REACTION) GIVES 6-10% HYDROGEN
- WELL BELOW DETONATION LIMIT
- DESIGN PRESSURE 60 PSI, FAILURE PRESSURE >120 PSI
- CAN ACCOMMODATE PRESSURE FROM HYDROGEN COMBUSTION
- POST-TMI IMPROVEMENTS REDUCE LIKELIHOOD OF ACCIDENT

STAFF POSITION ON HYDROGEN CONTROL

FOR LARGE DRY CONTAINMENTS

PENDING THE RULEMAKING PROCEEDINGS CALLED FOR IN ITEM II.B.8 OF THE TMI ACTION PLAN, NUREG-0660, THE STAFF IS NOT NOW IMPOSING ANY NEW REQUIREMENTS I.E., BEYOND 10 CFR 50.44, FOR HYDROGEN CONTROL IN LARGE DRY CONTAINMENTS. THE STAFF COMPLETED ITS REVIEWS FOR THE FULL POWER LICENSING OF NORTH ANNA 2, SALEM 2, FARLEY 2, AND TMI-1 (RESTART) WITHOUT IMPOSING ANY NEW REQUIREMENT FOR HYDROGEN CONTROL. THIS POSITION WAS BASED ON THE STAFF'S VIEW THAT LARGE DRY CONTAINMENTS HAD SUFFICIENT CAPABILITY TO ACCOMMODATE HYDROGEN COMBUSTION SO THAT NO NEW REQUIREMENTS NEEDED TO BE IMPOSED, PENDING THE RULEMAKING PROCEEDING ON DEGRADED CORE COOLING.

IN THE PROPOSED RULE PORTION OF THE INTERIM RULE, (SECY-81-245A), THE STAFF PROPOSES TO REQUIRE THAT CERTAIN SPECIFIC ANALYSES BE PERFORMED FOR ALL LARGE DRY CONTAINMENTS TO ASSURE THAT CONTAINMENT INTEGRITY AND SAFE SHUTDOWN WILL NOT BE JEOPARDIZED BY HYDROGEN RELEASES FROM THE POSTULATED DEGRADED CORE ACCIDENTS.

HYDROGEN CONTROL

J. O. Cermak, Manager of Nuclear Safety - SNUPPS

JS 11-5-81

HYDROGEN CONTROL SYSTEM

- REDUNDANT HYDROGEN RECOMBINERS
- REDUNDANT HYDROGEN MIXING SYSTEM
- REDUNDANT HYDROGEN MONITORING SUBSYSTEM
- BACKUP HYDROGEN PURGE SUBSYSTEM

SOURCES OF HYDROGEN GAS IN CONTAINMENT

* 5% ZIRCONIUM WATER REACTION POST LOCA
* POST LOCA RADIOLYTIC DECOMPOSITION OF WATER
* POST LOCA CORROSION OF METALS AND PAINTS

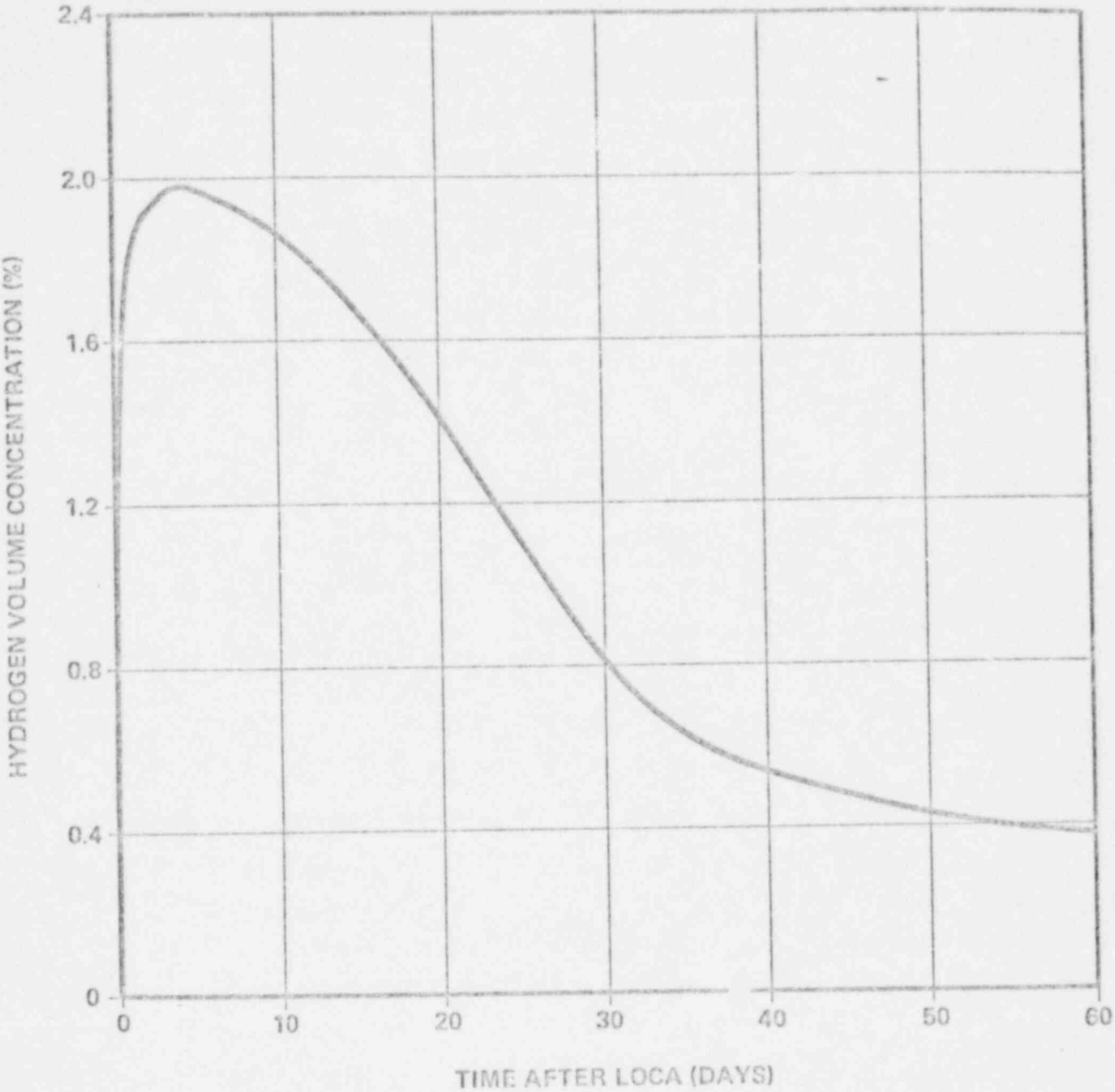


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FIGURE 6.2.5-2

HYDROGEN VOLUME CONCENTRATION IN CONTAINMENT WITH ONE RECOMBINER OPERATING AT ONE DAY

HYPOTHETICAL CASE

LARGE AMOUNTS

OF HYDROGEN IN

DRY CONTAINMENT

CONTAINMENT VOLUME - 2.5×10^6 FT^3

75% METAL WATER REACTION CASE

RESULTS

- 12.5% HYDROGEN BY VOLUME IN CONTAINMENT
- CONSTANT VOLUME AND ADIABATIC DEFLAGRATION OF HYDROGEN YIELDS PRESSURE INCREASE OF 60 PSI (75 PSIA)
- DESIGN PRESSURE OF CONTAINMENT - 75 PSIA

CONCLUSION

PEAK PRESSURE FROM HYDROGEN BURN EQUALS CONTAINMENT DESIGN PRESSURE

75% METAL WATER REACTION CASE WITH WORST CASE LOCA

ASSUMPTION

HYDROGEN BURN OCCURS AT TIME OF PEAK CONTAINMENT PRESSURE FROM LOCA (47.3 PSIG AT 134 SECONDS).

RESULTS

- PEAK PRESSURE IN CONTAINMENT IS APPROXIMATELY 125 PSIA.
- ULTIMATE STATIC FAILURE PRESSURE FOR CONTAINMENT IS APPROXIMATELY 175 PSIA.

CONCLUSION

FOR WORST CASE LOCA AND 75% METAL WATER REACTION, PEAK PRESSURE IN CONTAINMENT IS APPROXIMATELY 50 PSI BELOW ULTIMATE STATIC FAILURE PRESSURE.

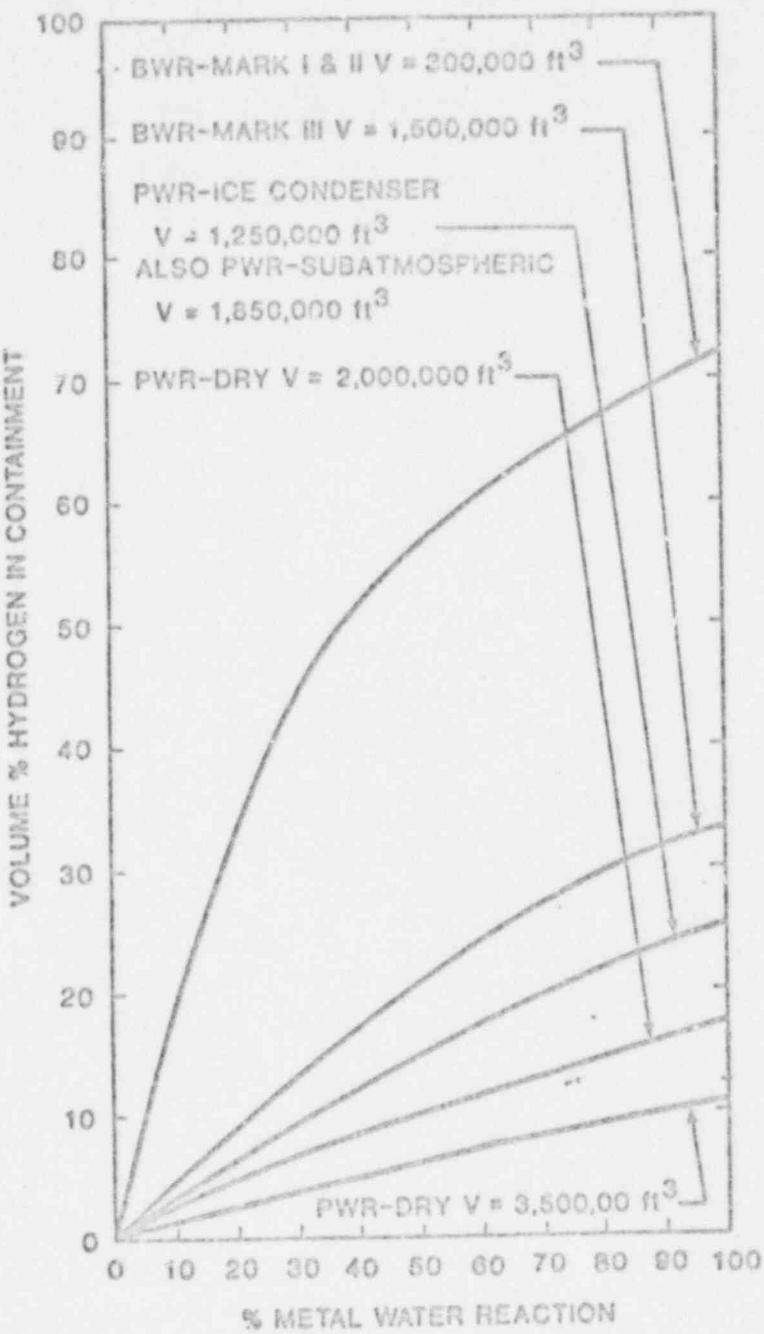

FIGURE 2
VOLUME % HYDROGE" IN CONTAINMENT
VS % METAL-WATER REACTION

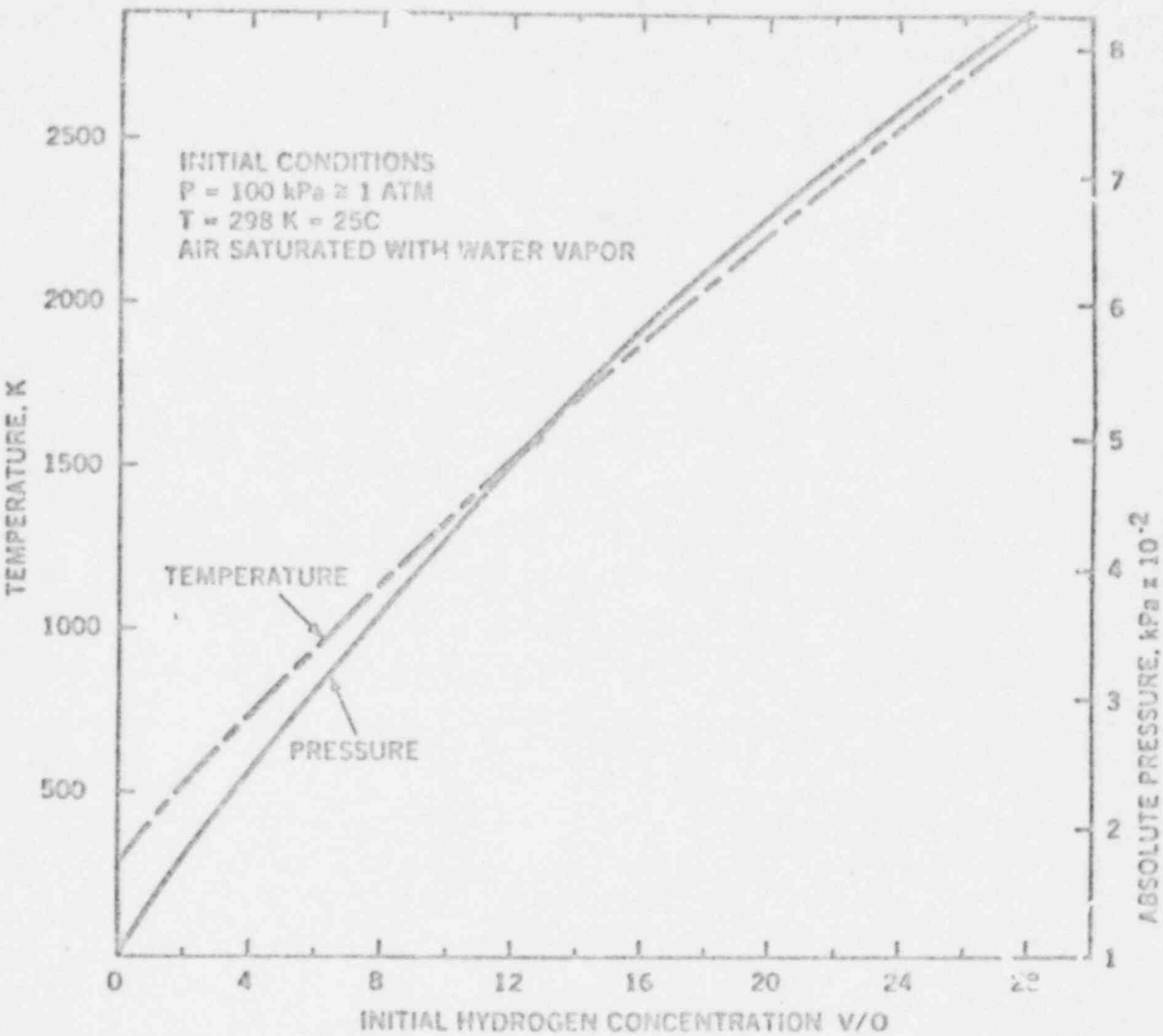

FIG. 3. PRESSURE AND TEMPERATURE AFTER HYDROGEN-AIR COMBUSTION, CONSTANT VOLUME AND ADIABATIC

DECAY HEAT REMOVAL SYSTEM

F. Schwoerer, Technical Director - SNUPPS

TG 11-5-81

FUNCTIONAL REQUIREMENTS FOR COLD SHUTDOWN

1. HEAT REMOVAL AT HOT STANDBY
2. BORATION OF RCS
3. COOLDOWN OF RCS
4. DEPRESSURIZATION OF RCS
5. RHR OPERATION