UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON SAFETY PHILOSOPHY, TECHNOLOGY AND CRITERIA

PROPOSED REQUIREMENTS FOR NTOL PLANTS

NEAR-TERM CONSTRUCTION PERMITS

Nuclear Regulatory Commission

Room 1167
1717 H Street, N. W.
Washington, D. C.

Tuesday, January 6, 1980

The subcommittee convened at 9:10 a.m., pursuant to notice, when were present:

ACRS MEMBERS PRESENT:

DAVID OKRENT, Chairman
WILLIAM M. MATHIS
JESSE EBERSOLE

DESIGNATED FEDERAL EMPLOYEE:

RICHARD SAVIO

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P R O C E E D I N G S

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

This is a meeting of the Advisory Committee on Reactor Safeguards Subcommittee on Safety Philosophy, Technology and Criteria.

I am David Okrent, the Subcommittee Chairman. The other ACRS members present today are Mr. William Mathis and Mr. Jesse Ebersole.

The purpose of this meeting is to discuss proposed requirements for NTCP plants near-term construction permits.

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

The rules for participation in today's meeting have been announced as part of the notice for this meeting previously published in the Federal Register of December 22, 1980.

A transcript of the meeting is being kept and will be made available by January 8, 1981. It is requested that each speaker first identify himself and speak with sufficient clarity and volume so that he can be readily heard.

We have received a number of written statements from members of the public. Copies will be included with

the transcript of this meeting.

It is my understanding that we have some members or representatives here from utilities or other industry groups who wish to make oral statements. My notes tell me that Mr. Walker from EMP and Mr. Myers from SCAGET which to do so. Are there others who have prepared oral statements that we don't know of?

(No response.)

MR. OKRENT: I guess not at the moment.

The agenda is planned roughly as follows: A brief executive session and then a discussion with the NRC staff on where they stand and what their proposed requirements for NTCP plants are. Following that dicussions with the NTCP plant owners. At that point we would take the oral statements from Mr. Walker and Mr. Myers and others if there are representatives and we would try to hold informal or back and forth discussions on the subject. Then the subcommittee might talk to itself or whatever later in the afternoon.

It is my understanding that the staff has just made available a draft document that is entitled "Licensing Requirements for Pending Applicants for Construction Permits and Manufacturing License" and I assume that they are going to tell us what is in it.

I have been given a suggested agenda in which

Mr. Purple would start off for the staff, followed by Mr. Ernst who will talk about reliability engineering, Mr. Muller on siting, Mr. Meyer on degraded core and Mr. Schwencer on 0719 changes.

If anybody wishes, there are copies of this part of the agenda outlined on a piece of paper and Dr. Savio can make these available.

Do we have any comments from members of the subcommittee at this time?

MR. EBERSOLE: I would just as soon get on with it. I have no comment.

MR. MATHIAS: The only thing is I think we need to get into the revisions to 0718, but that is on the agenda so we will hear about that later.

MR. OKRENT: Well, it seems like a good idea then to start and let the staff tell us where they have devolved to.

Let's see, now, I did I think ask Dr. Savio to advise the staff that we would very much appreciate hearing various aspects or pros and cons or various alternative approaches that they considered on each of these things and not just the final position.

Was that made clear to them?

MR. SAVIO: Yes, it was.

MR. OKRENT: So I assume that this will

automatically occur during the presentations; is that correct.

VOICE: That is correct.

MR. OKRENT: Very good, because I think that will make for a more helpful discussion.

Why don't we begin and Mr. Purple, I guess, is the first speaker for the staff.

MR. PURPLE: Thank you, Mr. Chairman.

I am Bob Purple from the Division of Licensing. The handout you just gave out I will put up on the viewgraph.

(Slide.)

I would like to spend about 10 minutes on my part with a reminder of where we are and how we got to where we are and a little bit of the background.

I will describe for you the essence of the public comments we have received on NUREG 0718 and give you a brief overview of how 0718 was revised. We will get into more detail on that a little later as an individual agenda item.

I would like to remind you that the staff started this effort to determine what the necessary and sufficient set of conditions or requirements should be for issuing new CP's in March of last year. It has been reviewed in its early stages with subcommittees of the ACRS and the ACRS itself in meetings in April, May and June and August.

In August the Commission approved the issuance of

NUREG 0718 for public comment. The comment period expired on November 24th and the staff has been busily engaged since then in responding to that comment and giving further consideration to what the requirements should be.

The revised NUREG 0718 is almost complete. The draft you have is the most recent version. We anticipate finalizing it this week for a presentation to the Commission next Tuesday, January the 13th, and, as I am sure you are aware, we are scheduled for the full committee this Friday to discuss this same topic. Since we are meeting with the Commission next Tuesday, I am sure they would appreciate any advice the ACRS could offer after we meet with them on Friday.

MR. OKRENT: Let's stay with that point a minute. What is the Commission on January 13th supposed to be? Is it supposed to be a presentation of a final position by the staff in this matter or what?

MR. PURPLE: With two exceptions, that is right. We will be seeking their approval for positions we will have taken, or all positions except two which I will discuss here in a little bit.

We may not have finalized a position on two issues. If we have not we would be laying alternatives in front of the Commission for their guidance, but it would be a decision meeting. We would be seeking their approval.

MR. OKRENT: Now, I am not sure whether the committee gets its written comments out by two days after a meeting necessarily, does it?

MR. SAVIO: In this case it could be out.

MR. OKRENT: It could be out you think?

MR. SAVIO: Yes, sir.

MR. OKRENT: Let me assume the committee in fact did prepare some comments. Wouldn't the staff need time to look at these before they met with the Commission, or do you already know it doesn't matter what the committee advises?

(Laughter.)

MR. PURPLE: I guess that rather depends on how profound and how deep the comments are from the committee. I mean obviously if they are extensive and attack greatly the position of the staff, I would imagine we would have to suggest to the Commission in the meantime to look at them.

MR. OKRENT: I see. All right, why don't we go on because I think the topic will come up again since I understand there are some people who have information that they may want to provide at a date later than January 13 that bears on the subject.

Go ahead.

MR. PURPLE: Again by way of review just to remind you how NUREG 0718 itself was composed or fabricated it is derived generally from the TMI Action Plan. In its initial

conception the idea was to take a look at what had been determined and the necessary lessons learned from the TMI-2 accident and consider them in the light of the new CP and from that lay out a list of requirements.

In the Commission paper in August that led to the approval to issuing this for comment we had discussed three options with the preferred option being one that I will characterize. It says you treat these existing CP applicants, the six CP applicants and the ML applicant generally the same as new OL's maybe with a little tightening of requirements and with special attention in four areas, the four areas being emergency preparedness, reliability engineering, degraded core rulemaking and the siting.

The Commission asked in addition that we seek public comment on the idea of requiring in addition that CP applicants compare their plants against the standard review plan. We did get comment on this topic for purposes of today's meeting and even for purposes of the Commission meeting next Tuesday. We don't plan to discuss it in the context of the NUREG 0718 because there was another effort by the staff to pick up not just the CP's for this effort of reviewing against the standard review plan but also for all OR's and OL's and CP holders. So they will all be lumped into that discussion and I don't propose and we are not

prepared to discuss that today.

With respect to the public comments received my best count was that we got 12 letters commenting on NUREG 0718, two from individuals, three from nuclear system suppliers, one from an architect-engineer, five from utilities and one federal agency that responded.

I will try to characterize the major comments as I see them with some trepidation because I may not pick some that some commenter thought was most significant. But generally speaking the comments that came in reflected a feeling that the requirements in 0718 went too far and would result in costly delays for these pending CP's and then urging on the part of most commenters to treat the CP's just like the OL's that are getting licenses today without the added special attention in special areas.

With respect to the siting issue the most common comment was pointing out that the Congress in its FY-'80 authorization directed the Commission to issue siting regultions but at the same time enjoined the Commission that these new siting regulations shall not apply to these particular CP's, that is, any CP that was docketed before a certain date. Those who made that comment felt that the things we were asking for with respect to siting went counter to that.

With respect to degraded core rulemaking in

general many of the comments were don't require anything now. Let the rulemaking run its course and treat these CP's as all other plants under construction or in operation would be treated at the end of that rulemaking process.

With respect to reliability engineering there was generally not a strong negative set of comments but there were some who commented that the methodology is not yet well established in the industry and we don't have any clear safety goals even if you had the methodology and that in general laying this requirement as we had it in 0718 is premature.

Briefly to tell you how 0718 is different now in your hands than it was when it was issued in August or September, first of all, no new items were added. No new action item plan item or any other item was added to the list.

We rewrote the text of many of the items. In strengthing the requirement I was able to find five items and in clarifying and updating because time had passed by and certain events had happened in eight or ten others.

As you may recall, what we did for each action item in 0718 is they were assigned to a category one through five, category one being at one extreme that says it is not applicable to CP's so that action plan item doesn't apply and has no meaning, and progressively more stringent

requirements up through category five meaning the one for which the applicant would be required to provide a full description and a full completion of the item, whatever it calls for, at the time of the CP or the PSAR submittal.

Now, in those terms we changed the category designation of a number of items. The numbers of items I am a little reluctant to get into because it varies if you are talking about a construction permit or a manufacturing license.

Let's take, for example, the construction permits. We change the category designation of some 23 items out of a total of 95 or so that were in the document, 16 of them being redesignated as category one. That is a decision made that after all this item is not applicable to a CP review and need not be considered further.

Four were upgraded to a higher category; that is, more detail would be required and more analysis by the staff prior to issuance of the CP and three were downgraded to a lower category, other than the 16 that were downgraded all the way to category one.

With respect to the four special areas that were called out for special attention in NUREG 9718, the emergency preparedness, as I think you know, the final rule on emergency preparedness changed to Part 50 in Appendix E is now final and in place. So there is really no longer any

speculation on what the requirement here is. It is clear with respect to these CP's and would simply follow the regulation in that area.

With respect to reliability engineering risk assessments, we have had considerable discussion on this within the staff. The bottom line is we now are preparing this item. It would still require what we call a simplified reliability engineering anaysis to be performed on selected systems that were identified in the requirement, but we have noted in the statement of the requirement that it is very likely that as time goes on that we are going to be requiring in a broader program that full risk assessment be performed on all these plans as well as all other plans. At such time as that is laid on as a requirement it might replace and supersede the need for doing these simplfied reliability analyses.

Now, Mel Ernst, whom I haven't seen yet this morning, will speak to this in more detail in a little while.

With respect to the siting issue what we had in NUREG 0718 in the draft as it went out was a requirement that licensees compare their sites against the Siting Policy Task Force Report, NUREG 0625 as amended by the ACRS comments and as amended by the Office of Policy Evaluation comments so that the staff could determine and the Commission determine whether or not there were deficiencies

with respect to those criteria that would lead to the possibility for plant modifications to improve or to ameliorate what would be an unsatisfactory condition.

One of the things that made this particular issue more complicated was that as we issued the draft 0718 we thought at that time that the proposed new siting rule would be on the street by October of last year. There has been a slip in that schedule. It is not scheduled now to be issued until April of '81, the proposed rule.

So at this time we don't really have a clear set of criteria. It simply referred to NUREG 0625 as amended by comments from the ACRS and OPE. It does not provide a very clear set of criteria. You really can't tell when you take those three bodies of documents in any particular area what criteria you are supposed to use.

Leaving the description of the siting issue as it was, tha[illegible] comparison against this 0625 as modified, we believe would lead to very difficult litigation in the hearing process because of the lack of clear criteria. For that reason we are considering a range of options trying to develop a more precise set of criteria against which to c[illegible]pare these plants. Mr. Muller will discuss this a little later in the agenda.

Finally, and perhaps the most difficult of all, is the degraded core considerations. The draft 0718 language

in this area is considered now to really be too vague and would lead to very difficult litigation in the hearing process. It would essentially leave to the Board to determine exactly what the staff had in mind. We feel it important that we try again as in the siting rule to develop a more precise and understandable set of criteria of what we are really looking for in this area.

Again, we have developed a range of options. Jim Meyer from the Division of Systems Integration will discuss that range.

I mentioned, when you asked me about the Commission meeting next Tuesday, what type of meeting it was. I said it was a decision meeting where the staff has taken a position on the entire package with the exception of two items. Unless we do miraculous things for the remainder of this week, the two items would be the siting issue, for which we expect to present to the Commission the range of alternatives, and the degraded core considerations, again for which we expect to present a range of options.

Mel Ernst's timing is superb.

(Laughter.)

That concludes the introductory overview remarks. What I would suggest is either if you have some general questions that I can try to handle, we can do that, or we can move right on to the reliability engineering and our

approach to that.

One disadvantage I guess, Mel, is I don't know if you heard what I said.

MR. ERNST: I apolgize for being a little late but I got tied up.

MR. PURPLE: May I take that out of order and, if you don't mind, could we talk about the siting right now and give me a chance to let Mel at least hear what I had to say about his topic before he discusses it?

MR. OKRENT: It is all right with us.

(Slide.)

MR. MULLER: I am Dan Muller of the Nuclear Regulatory Commission staff.

Bob substantially pointed out the range of options that we have with regard to siting. One would be to just go ahead with the siting part of these CP reviews really ignoring what is going on at the present time with regard to siting. In other words, treat them under the old rule 10 C.F.R. Part 100 and Reg. Guide 4.7 if that happens to be applicable.

The other option on the other end of the spectrum would be to stop the CP reviews of these plants at this time pending the availability of either a proposed rule or a final rule on the siting which would amount to some fairly significant delay.

The third option which we are going to propose ---

MR. OKRENT: Will I find these option in this draft 0718?

VOICE: No, they are not in there.

MR. MULLER: The third option is where the CP's and the manufacturing licenses will be issued for those applicants that meet the requirements of our current siting regulation Part 100 and also if necessary as specified in Reg. Guide 4.7. However, after the proposed siting rule is on the streets then we will have the applicants review their sites against the new rule. This will not be a condition of the CP, however. CP's may be issued in the meantime.

Then finally the staff will evaluate the combined site and plant characteristics and decide whether any design modifications are recommended, but this is, as I said, subsequent to the issuance of a CP.

MR. OKRENT: Let's see, the first option again was?

MR. MULLER: The first option would be to go ahead and license and effectively grandfather -- I guess that is the right word -- all of the current CP's that have been applied for at the present time and not do the second two parts.

MR. OKRENT: What criteria would the staff use for part 3 of the alternative you have on the screen?

MR. PURPLE: The last part, deciding whether any

design modifications are appropriate, Bob.

I think as we see moving forward on this that the combination of the siting issue, the degraded core considerations and the reliability engineering assessments, risk assessments, will lead to eventually in their combination a generalized risk assessment for each of these plants.

We think based on that that one would be able to determine if there are some very bad spots with respect to site characteristics what the worth of various modifications might be. I think we would probably end up using an approach similar to that which is now being used in the Zion and Indian Point review which is generally a siting problem and we are doing a risk analysis there to determine what kind of fixes are needed. I am pretty sure we don't have a set today of clear yardsticks that we would use to measure that.

MR. OKRENT: Do you think there are criteria that exist for the Zion and Indian Point review?

MR. PURPLE: That will probably be a milestone that will establish what criteria we will use in the future.

MR. MULLER: It may help to answer your question if I can go into another part here.

MR. OKRENT: Is this something that satisfies the lawyers?

MR. REIS: Yes. It is a continuation of part

practice where we have looked at the location of the plant and evaluated it against what design was needed for the plant. In other words, there were places where we had demographic features that required increased safety components to the plant and this is a continuation essentially of that. We are coming up with new siting requirements. However, what we are applying here are the safety features and the engineering features to the plant and not ruling out any sites. We think that within that context this satisfies it.

MR. OKRENT: I am not sure whether the lawyers have a good picture of when it is the proper time to make design modifications. Legally you can make them any time.

MR. PURPLE: I think if Dan goes on what he is going to show you, and I think you have in the handout as well that we handed out earlier, is a comparison of these sites against the criteria I guess in 0625 as well as Dan probably has some feel for what they would likely be in the proposed rule. I don't think this is going to be a major problem for any of these sites. So it may not be a real problem.

MR. OKRENT: All right.

(Slide.)

MR. MULLER: These are the six near-term construction permit sites compared against the NUREG 0625

figures. Across the top are the NUREG 0625 figures, zero to five miles, 100 per square mile, five to ten, 150 and ten to twenty, 400 per square mile, or in the case of zero to five a half of the regional population density, .75 of the regional population density or twice the regional population density.

Then from that one can come up with, for instance, a hundred per square mile is equivalent to 3,925 people within zero to five miles.

Then I compare that upper limit population with the populations for the specific plants. It is lower than that except for Pilgrim.

However, then, if one goes into the regional population density approach you find very quickly that you can accommodate 14,000 in the region and the actual population is 5,342.

MR. OKRENT: What do you mean by accommodate?

MR. MULLER: If I take the regional population density and consider that the state in which Pilgrim is located, then I have a population density for that state. I take half of that population density and say that is the allowable population density around the Pilgrim site. Then I compare that with the actual population density and I find out that the allowable population density is somewhat lower than the actual population density.

MR. EBERSOLE: Are there any compensatory aspects to excessive population density in the context of improving the mobility? I don't regard population density as a parameter of particular interest unless it is associated with mobility of lack of it.

MR. MULLER: Well, it is very important obviously in the evacuation capability. The studies that I have seen on evacuation capability for a number of existing sites would indicate that there really isn't a marked difference in the mobility or the time for evacuation from site to site?

MR. EBERSOLE: Could they be enhanced?

MR. MULLER: I suppose they could be enhanced if one were to go to let me say heroic measures to notify people and then ensure that those people that are indeed notified have the transportation capability to get away. By heoric I suppose you could do things like have, you know, a radio or something specifically tuned in in every residence. You still run into problems where the kids are out playing or the farmer is out plowing or something of that sort.

MR. OKRENT: I would like to understand what I see on the slide.

MR. MULLER: All right.

MR. OKRENT: At Pilgrim the 363 number in brakets is?

MR. MULLER: There are 122 people per square mile at the Pilgrim site, but if one considers the regional population density which is the state in which Pilgrim is located divided by two, because it is half of the regional population density, then the allowable population density is 363. Compared to 122 obviously it is higher than the 122.

Then on the basis of the way we were looking at NUREG 1625 we said that site would have been acceptable assuming the region is the state. There are a number of assumptions on it.

MR. OKRENT: Those were your assumptions?

MR. MULLER: Yes.

MR. OKRENT: Are you assuming that that is the region?

MR. MULLER: I am assuming that the region is the state just for the purposes of this table.

MR. OKRENT: I don't know what it means to say for the purposes of this table. Either you are trying to tell us something by these numbers or not.

MR. MULLER: I am really answering your question in terms of you asked us to compare these six sites with the illustrative examples in NUREG 0625 and that is what I have done here.

What I would like to do is go on because what we are finding as we have gone through a considerable amount of

study of what the population densities might be in the vicinity of these plants is we are finding we can be considerably less restrictive. So these illustrative examples of NUREG 0625 are much more restrictive than we think now we will be recommending in terms of population distribution.

MR. OKRENT: Well, why don't you go on.

MR. MATHIS: May I ask one question first?

MR. OKRENT: Sure.

MR. MATHIS: If you included Rhode Island in your number for Pilgram rather than just Massachusetts would that appreciably change that number?

MR. MULLER: No. Do you have a handle on it, Len?

MR. SOFFER: I am Leonard Soffer of the Siting Analysis Branch. I don't know precisely what the population density in the area of Rhode Island is but it is not very different from Massachusetts and I would estimate that it wouldn't change significantly.

MR. MULLER: I guess Connecticut, Massachusetts and Rhode Island are fairly comparable, are they not?

MR. SOFFER: That is correct, they are.

MR. MATHIS: On an average basis.

MR. SOFFER: On an average basis.

MR. MATHIS: But on a localized basis there is terrific difference. That is what I was trying to get at.

MR. MULLER: If one takes all of New England then that makes quite a difference, too, because you get those areas of Maine where there is very little population density.

We have been working now toward developing specific siting population criteria. What I will give you now are somewhat preliminary thoughts but we are beginning to focus in on what these population densities might be. We have been guided in developing these numbers very much by your recent publication on proposed risk from the presence of nuclear plants.

It looks like this fine structure that we have from zero to five, five to ten and ten to twenty probably isn't necessary. It looks like we can even likely get away from a regional approach. Profoundly enough it seems that the number is coming in at around 500 per square mile out to 30 miles which isn't all that different obviously from Reg. Guide 4.7.

It appears however that we can accommodate some fairly significant improvement in the number of prompt fatalities for a large accident if we were to be more restrictive in very close to the plant, and by close I mean something like within two miles or so. So we are thinking now of some fairly restrictive population density within two miles and then 500 per square mile from two to thirty.

If one calculates the risk to individuals from

this large accident we find that we can be well within the recommended risk criteria that you have recommended in your report. In fact, we are like an order of magnitude below that.

MR. OKRENT: That is assuming something.

MR. MULLER: This is assuming the accident that you proposed. Your risk goal was ten to the minus four probability of a large accident. I am fuzzy on the number.

MR. SOFFER: What we have done is we have asked Sandia and Dames and Moore to do some studies for us and these studies are still continuing. Basically we have asked them to look at the consequences of large atmospheric releases. We have asked Dames and Moore to looking at siting availability studies. We have been guided by the safety goals that are in NUREG 0739.

Using an assumption that a core melt has a frequency of about one times ten to the minus four and that a large atmospheric release has a probability of about one times ten to the minus five it appears that we could meet the safety goal that the ACRS has tentatively identified in NUREG 0739 for a maximally exposed individual at approximately an exclusion radius of about a half a mile.

Furthermore, it appears that the societal goals can be met with population densities on the order of several thousand people per square mile.

Consequently we believe that the criteria that we will be proposing will be well below the numbers that have tentatively been identified by the ACRS, except we believe it might also be prudent, as Dan has mentioned, to try to do something in the immediate vicinity of the reactor where it looks like key fatalities are an important consideration. Consequently we are looking at somewhat more restrictive numbers in that immediate area.

MR. MULLER: As I recall, it is like if one calculates it about two-thirds of the acute fatalities occur within two miles and roughly another third occurs out to five miles. So if one were to decrease the population density within two miles you can decrease the number of acute fatalities in proportion. That is the type of thinking we are working on right now.

MR. OKRENT: I guess I would have to see something written down before I could absorb what you have just told me.

In the first place, I would like to caution you against taking those numbers as firm recommendations. They were put out as a step in what was expected to be an interim process and they were supposed to be a point of discussion to see if in fact were these useful, plausible or whatever.

Secondly, I would like to advise that you find out. In fact, even if you think they are all right, you had

better find out what does the Congress think, for example, before you plan your long-term safety policy around that or any other set of numbers.

Thirdly, and most importantly, I would like to warn you or anybody else that it is not intended that these be used piecemeal. You don't take a part of that set of recommendations that happens to be useful for what you are doing.

For example, there was also in that set of recommendations some conditional requirements that given a serious accident to the core there would be a low probability of a release. Have you looked to see whether the reactors that you are taking about would meet that conditional requirement? Because if they don't meet that conditional requirement they are violating part of the criteria.

The whole principle there is it is not enough to meet one, you are supposed to meet all the criteria. So any single one perhaps was not made as rigid as it might be if it were the only one to be met since the intent was that if you met all of these then you would have what the staff sometimes calls multiple barriers of defense and there wasn't an intent that way and there was still an ALARA criteria.

Anyway, I am a little bit surprised to hear you

say that these lead you to a conclusion that you could get, what was the number you said, 5,000? I think you said thousands.

MR. MULLER: Let Len go through it.

MR. SOFFER: These numbers are a little surprising to us as well. What we have done is look at the demographic implications and the societal health risk limits, that is the ones that are listed in Table 3 of NUREG 0739. It appears, first of all, that the acute fatalities or early deaths are the more limiting then the latent cancers with respect to demographic criteria.

MR. OKRENT: To the individual.

MR. SOFFER: As far as meeting the goal. That is, the early deaths require lower demographic values, lower population density numbers than the latent cancer numbers.

MR. OKRENT: Go ahead.

MR. SOFFER: If you accept that the probability of a core melt with a large atmospheric release is one times ten to the minus six per year and are using the expected value of .4 deaths, that is the goal value ---

MR. OKRENT: I am sorry, what is ten to the minus six per reactor year?

MR. SOFFER: That is the probability of a core melt with a large atmospheric release.

MR. OKRENT: Are you accepting that?

MR. SOFFER: I am not accepting or accepting. I am just saying I am going to give you three sets of numbers and talk about them.

MR. OKRENT: All right. Go ahead.

MR. SOFFER: If you assume that that number is one times ten to the minus six, and if you assume that the expected number of early deaths to be achieved is 0.4, then that can be met according to our calculations using the formula as given in NUREG 0739 with a population density of 38,000 per square mile uniformly.

MR. OKRENT: A problem that arises in the set of numbers you just indicated is that there is another memorandum that exists with the staff which has estimated that the probability of a serious accident which would lead to a large release for a certain number of plants may be much larger than ten to the minus six.

MR. SOFFER: Yes, indeed.

MR. OKRENT: So you have to be a little careful about planning a siting around numbers that you don't think you are achieving and which you don't at the moment have designs to achieve perhaps.

MR. SOFFER: That is quite correct. Let me say the staff certainly has no intention of proposing sites with 38,000 people per square mile seeing that the Island of Manhattan only has a density of 26,000 people per square

mile.

Let me give you another calculation. If you assume that the probability of a core melt with a large atmospheric release is five times ten to the minus five per year, and if you take the upper non-acceptance limit of two early deaths, then using the formula given in NUREG 0739 that can be achieved with a uniform population density of 5,600 people per square mile.

Finally let me give you the hybrid case. Let me assume that the probability of a core melt with a large atmospheric release is five times ten to the minus five but now you wish to achieve the lower limit on early deaths of 0.4 that I mentioned before then that implies a population density of about 1,500 people per square mile.

What we have done is we have merely looked at these to give us some sort of insight and give us a feel for where we are.

MR. OKRENT: Well, I think that is interesting. Of course, when the committee commented on your Siting Task Force report it I think made some comments with regard to population density. I don't have the words handy, but they were to the effect that one wouldn't use more populated sites if one didn't need to, for example, and I think it reiterated some of its earlier comments of many years ago about not putting plants near to large population centers

and so forth.

So, again, I think one wants to not take a part of the recommendations and use that. I think in fact what you would find if you took the approach you have just said and you were to take a typical reactor and site it around these large populations that you have just described, when you put the ALARA condition on in 0739 you would find that it was cost effective to make lots and lots and lots of improvements because you computed big effects possibly.

That is part of the package. In other words, you don't have an ALARA that you really apply, and I am not saying I know how you would apply it. Then you have to rethink how you approach the same question.

Go ahead.

MR. MULLER: That is substantially it. I am finished.

MR. OKRENT: Could you flash this figure back on the screen.

(Slide.)

Now, on the top it mentions "Illustrative Criteria in NUREG 0625."

MR. MULLER: Yes.

MR. OKRENT: Now, I have the impression that many people, and this is people in the U. S. and outside of the U. S., seem somehow to have thought that these were not just

illustrative criteria but they were recommendations by the staff.

MR. MULLER: That is right.

MR. OKRENT: Haven't you read that into what various people have said?

MR. MULLER: I have read that into what practically everyone has said and I am frustrated because evidentally those who say that don't read the report.

MR. OKRENT: The ACRS in fact when it commented on your report chose to comment only on the general recommendations and took your word that everything else was just there for purposes of discussion.

MR. MULLER: For purposes of illustration, not even discussion.

MR. OKRENT: Well, it was marked discussion and not recommendation.

MR. MULLER: Yes.

MR. OKRENT: So in fact those are not the recommendations of the staff?

MR. MULLER: That is right.

MR. OKRENT: They are illustrative?

MR. MULLER: That is right. They were put in so that those who were reading the report might be better able to understand the recommendations.

MR. OKRENT: If that is the case, what is it that

these aTCP sites were supposed to compare against?

MR. MULLER: Compare against the new requirements that will be coming out for siting. They will compare against the new proposed rule which is something that is coming out in April.

MR. OKRENT: I see. It is still not available.

MR. MULLER: That is right. You see, that is why I am suggesting, too, that if the new proposed rule has figures in the range of 500 per square mile out to 30 miles then all of these sites will very readily meet that.

MR. OKRENT: The Congress I thought had asked -- and I can't remember the exact wording -- but that the Commission tend toward more remote siting. Am I wrong?

VOICE: No.

MR. OKRENT: What was it that you have asked by the Congress?

MR. MULLER: I have it right here.

MR. OKRENT: You were asked to develop siting requirements independent of design.

MR. MULLER: Yes. It says "Regulations promulgated pursuant to this section shall specify demographic criteria for facility siting, including maximum population density and population distribution for zones surrounding the facility without regard to any design, engineering or other differences among such facilities."

MR. OKRENT: Is there a legislative history on that bill?

MR. MULLER: Yes.

MR. OKRENT: Does it give any more insight?

MR. MULLER: I am sure it does.

MR. REIS: Yes, it does. Generally the tenor of the legislative history or at least of the Conference Committee report was that this was to be, as is indicated in the language itself, independent of the design of the facility. There should be demographic limits.

Now, the bill also said that the regulations as such do not apply to the plants where construction permits applied for prior to October 1st, 1979, which are all these.

However, we are essentially looking at what we would get and seeing whether other changes should be made in the design of the plant which were recognized prior to that.

MR. SCHWENCER: In effect for new sites future looking you are not supposed to take into account mitigating capabilities. The site is supposed to be on the basis of whatever these new rules are proposed to be. But for this transition period we have a number of plants out there perating under construction and these current CP applications where the law apparently would say you can't apply this new criteria but it doesn't say you can't look in terms of the old past practice of mitigating features where

you think this would make a significant improvement in safety.

MR. REIS: In other words, the law limits setting demographic requirements for the siting of the facility itself. It does not prohibit looking at the demographics of the situation where you plan to build the site and seeing what engineering features the plant needs and what design features the plant needs so that it may be sited at that location which has been the past practice of the Commission.

MR. EBERSOLE: That old mitigating concept was a highly stylized thing, mitigating consequences of a principal LOCA. Is there anything afoot about adding additional preventive features which is under consideration? As we now well know the principal risk of plants is what we call loosely transients and not LOCAs.

MR. SCHWENCER: Dan, do you have that option slide you used?

(Slide.)

MR. SCHWENCER: I think basically we would be looking both mitigative and preventive in terms of whether or not there are modifications, either design or operations, of the facility. That would be preventive and mitigative. We are not being restrictive in our thinking as to which of these would go. I know in terms of Zion and Indian Point risk studies both mitigative and preventive are under

consideration there and I don't think we would change our thinking.

MR. OKRENT: Could you put the previous one back on once more.

(Slide.)

MR. OKRENT: Under the column "Maximum Sector Total" for Pilgrim you show 5,342. What is that angle in that sector? Could you remind me? Is that 22 and a half?

MR. MULLER: Twenty-two and a half.

MR. OKRENT: How many people are there within the first five miles at Pilgrim total?

MR. MULLER: It is 122 times whatever the area is.

MR. SOFFER: About 80 square miles.

MR. OKRENT: 10,000?

MR. SOFFER: Approximately.

MR. MULLER: In fact, I think it is half the sector so it would be twice this 5,342.

MR. OKRENT: Now, you said if we compare it against half the density of the State of Massachusetts it is less than that. Now, are you seriously proposing using state lines as the definition?

MR. MULLER: At the present time it appears from what we have been doing it looks like we might be able to not have to use this regional approach at all but rather for the entire country use something like 500 per square mile.

The only possibility of a regional approach may be in setting these very close limits when we are talking about a limit between zero and two miles. Possibly then we may go to a regional approach where we would make one region generally the northeastern part of the country where there are relatively few sites that are available that have a sufficiently low population density. As soon as you get into the Midwest or southern part of the country or the West the population density is quite low and there is no real problem in siting.

MR. OKRENT: We are supposed to be talking about NTCP sites and it is a little bit hard to divorce it from the other.

Let me ask you another question. Do any of these sites afford let's say what you would call serious questions with regard to groundwater or this sort of thing in the event of a serious accident or are they all good sites in that regard?

MR. MULLER: Someone is supposed to jump up and answer that question?

(Laughter.)

MR. BIVINS: My name is Bill Bivins. I am the jumper-upper.

(Laughter.)

We have made an assessment as opposed to an

evaluation of these sites and there are no characteristics on these sites which leap out and suggest unusual problems at any of the particular sites. I would stress we have not gone through one of these sites and done a detailed re-evaluation of the groundwater, but there is nothing in the current evaluation that suggests any particular problem with these sites.

MR. OKRENT: Now, there is a draft Sandia report, I think it is still a draft, which purportedly looked at the existing sites and said about half of these are really very good sites and that even if you had a serious accident and a lot of radioactivity got into the ground it wouldn't move to drinking water in any time period in which you were concerned. Do these all fall into that, I will call it, desirable category? Can you say that?

MR. BIVINS: I once appeared before you and found myself in the unenviable position of defending that particular report that you just made reference to. I swore at that time I would never do that again.

(Laughter.)

That report has not been finalized and has not been endorsed by the resource agency and to my knowledge no one at NRC has accepted the information in that report.

However, to try to answer your question ---

(Laughter.)

- these sites display, to the best of my knowledge, groundwater characteristics which would essentially make the chances of serious consequences through the groundwater pathway approach zero. None of them are significantly bad that they should fall, to my knowledge, into the category which the researcher at Sandia attributed to them.

MR. LEAR: That is based upon the time that is available for getting in there and doing some mitigation like slurry walls and pumping out of the groundwater. There were several in that very quick survey that was done last summer which were identified as having groundwater passage time in the order of months from the site to an aquifer that is being used for drinking purposes or a river that is leading to a drinking source. You have to get in and so something rather than taking a passive approach.

MR. OKRENT: You have to get in and do something for which sites, the good sites or the not-so-good sites?

MR. LEAR: The bad sites. Among those that I labeled poor from the viewpoint of doing something about it are Perkins, Pilgrim and Pebble Springs. Those three again, as I said at the outset, are based upon the quick evaluation last summer of the time it takes to go from the site itself to a drinking water source of some type.

The next step ultimately would be a full-blown

review of the type that is being done at Zion and Indian Point to see what would happen if you had no mitigative action and the fluid was allowed to go to that drinking water source.

MR. OKRENT: Are you talking about a drinking water source for a few people, a few hundred people or a few thousand people, ten thousand people?

MR. LEAR: That is site dependent, of course.

MR. OKRENT: But on the sites you mentioned. You mentioned two on the board there.

MR. LEAR: Well, several of them are on the Columbia River. I don't know offhand how many communities draw water from the Columbia River. However, in those particular instances it is a long time between the site and the river itself to travel in the ground.

MR. OKRENT: You said though that there was a period of only some months.

MR. LEAR: Yes, for Perkins, Pilgrim and Pebble Springs.

MR. SCHWENCER: It seems to me I saw something, George, that at Pilgrim there would be no potable water.

MR. BIVINS: Let me speak again because I think George is talking about groundwater travel time as the key here. In fact, the case which is mentioned, Pilgrim, which has the shortest travel time, the transport is to Cape Cod

Bay which isn't used for potable water. I think we may be talking to two different issues here.

MR. OKRENT: Is there a consideration like that for the FMP though for Pilgrim then for seafood? Would that be significantly affected.

MR. LEAR: We have not as yet done that type of an analysis. That has been our approach on some selected plants that up for OL's right now. We do an analysis of the site to determine the effect on the environment under certain assumed source terms and compare it to the FMP with the pathway analysis to get a relative magnitude of the effects but we have not done that as yet.

MR. OKRENT: Maybe we can come back to the first viewgraph that gave your preferred alternative. Now again could you tell me what you think No. 3 means and if I were an NTCP owner how I would interpret this and try to figure out what is going to happen or however you want to put it?

(Slide.)

MR. MULLER: I think to repeat somewhat I guess what Bob said, we are going through this process at the present time for Indian Point, Zion and now Limerick of relooking at the design of the plant vis-a-vis the siting and will likely continue to do this for some other plants and we would pick up these in the same manner.

So if you ask me to give a specific criteria or a

basis for what we are doing I guess I can't give you a straight answer at the present time. But I see this really as a continuation of this process that we are currently doing for Indian Point, Zion and Limerick. I think that one has to keep in the back of one's mind that except possibly for Pilgrim all these sites are relatively low population density sites. They fall probably well below the average sites that we currently have licensed.

MR. OKRENT: Does the subcommittee think they know what this means so they could explain it to the full committee?

(Laughter.)

MR. VOLLMER: I hate to see you go to the full committee with that impression

(Laughter.)

I think what Dan and Bob were trying to say, at least as I heard it, and I thought they did, was two things. We feel at this point in time, based on what we think we will be recommending as interim criteria for the proposed rule on siting, that it doesn't look like these sites will be outliers in terms of population density or other criteria that we will be proposing for the siting rule.

However, if there are certain features of these sites which do not fall within the criteria proposed for the

interim rule we would then want the licensees to go back and take a look at the deviations and the implications of these deviations for their particular plant and give the staff their evaluation of their plant vis-a-vis the proposed rule. If we found anything that really stuck out we would then handle it in the same fashion that we handled Zion and Indian Point and other plants.

We feel for a couple of reasons, one, that we feel the plants will fall within the proposed criteria and, secondly, with the implication of the law as we read it to perceive the CP's, we think it is prudent and reasonable for us to go forward on the CP's.

If something has to be done to the plants from an engineering standpoint because of an undesirable site feature, we feel it is reasonable to postpone that until after the CP is issued unless the CP drags out for an amount of time and the siting rule becomes ineffective or something like that. Ultimately there may be some specific grandfathering and the specter would be the site rule itself which will perhaps deal with these plants in specifics

MR. OKRENT: Mr. Mathis.

MR. MATHIS: Well, the licensees have been particularly critical of the lack of stability in the overall regulatory process. Here is another case of where it seems to me we are starting off with all kinds of if's

and and's and assumptions that basically are quicksand. How you can come up with this and derive anything that provides any stability to the licensee who wants to enter into a construction permit just escapes me at the moment.

I know you would like to hear something more positive, but I am just trying to give you an opinion so far and I feel like it is quicksand.

MR. SCHWENZER: Al Schwenzer of the Staff again.

I think one of the elements of stability is that we are saying site determination essentially is based on existing criteria, existing rules, and that the additional things that I think we have acknowledged are still future tense; that one, two, three iteration policy that we would intend to present to the Commission would in effect say at some later time, when these criteria, NUREG 0675, or whatever it turns out to be in the proposed rule, when those are available, then and only then would the licensees or the applicants in these cases be asked to address those things. So there was a conscious effort to try to create some stability in allowing them to proceed, consistent with the FY '80 Authorization Act, to proceed with the sites themselves.

So that,I would think if I were an Applicant, would be a very significant thing. We are not imposing site moving kinds of things here. It's in the framework of preventive or mitigative things for the existing site. That has a lot of ramifications for a plant, for example, like Black Fox that has an LWA, LW-1. The work, I understand, is essentially completed. I would think that would be fairly comforting to know that we are not looking at some new rules as a second look to see whether or not the site should be literally moved. We are proceeding on the basis -- we would be proceeding on the basis of the existing criteria and requirements as it indicated on that first slide.

MR. OKRENT: Okay. Why don't we go on and see how it goes.

MR. PURPLE: Now I'd like to move on then back to where the agenda would have called for reliability engineering topic.

MR. ERNST: I don't have any slides or anything prepared with regard to performing reliability risk assessments on NTCPs. I think the comments I might have would also perhaps apply, at least in some regard, to plants under construction at the present time.

I think everyone, including the industry, is desirous of performing some kind of reliability risk assessments. I think the big question is what methodology and when.

The question of criteria, what we would do with the assessments once we have them, is an excellent question. I guess it is the Staff's view that while we want to move ahead rather rapidly in the area of risk assessment and reliability assessment, we also want to move ahead hopefully in as disciplined a way as we can, with some criteria in mind, and some methodology in mind.

In that regard, I guess we picked the option for NTCPs that is expressed in the original proposal, but modified somewhat based on the Staff considerations and public comment received.

I guess there are three ways one could go for NTCPs. One could say we don't have the methodology; we don't a safety goal, therefore let's not do any[illegible] at the present time. I

guess that option is rejected by Staff, feeling that we have to move forward in some kind of a reasonable manner.

The second option would be to perform a full-blown risk assessment on all plants, all NTCPs, at some time, certainly perhaps prior to OL licensing, and that is an option.

There is some problems with that kind of an option: Number one, with reasonably standard methodology being developed in a timely way to apply it; and secondly, Staff believes that there is a constraint on resources, both within the Staff, as well as within the industry to perform such assessments. Therefore, there's going to have to be some priorities.

Priorities might include population density. However, after listening to this presentation, one wonders how much that might weigh. I don't know.

Differences in design might be one factor to consider in establishing priority, and the fact that a plant is in a licensing phase might be another consideration priority, but we have not really straightened out priorities. There's up to a hundred or more plants identified operating and in the licensing process to consider from a priority standpoint, to allocate resources effectively. So I guess we are very reluctant to state that we would require a full-blown risk assessment for NTCPs.

Therefore, the proposal that Staff is making is that as a minimum, we require some kind of a reliability analysis of critical systems, and we have an option open that based on

priority judgments, availability of a reasonably agreed-upon methodology, and useful allocation of resources, that we modify the rule slightly to allow the -- my hope is a fairly high probability fact that you might indeed do a full-blown risk assessment on each of these plants at some time during the construction and OL licensing review phase.

So that is basically what the Staff position is right now, that we at least require a simplified reliability assessment on critical systems to get some feel of weaknesses in the system and compare them to other studies and at other plants to see how they stack up against other studies, but have the option remaining to require a larger scale reliability, or perhaps risk assessment, depending on the circumstances in the future.

MR. OKRENT: Now what is the intent of this requirement? What do you think it will accomplish?

MR. ERNEST: I guess the intent is that experience, I guess, has demonstrated that there are nuances in design and improvements that can be made in the design, if one looks closely enough at it, in a reliability sense, that improvements can be made, and why not during the essentially final design construction phase, but before licensing for operation, why not require this kind of reliability assessment be done by licensees as a minimum.

MR. OKRENT: But there's nothing in the words that asks the applicants to modify their design. It asks them to do

a reliability -- simplified reliability assessment or something like that.

MR. ERNEST: That's correct. We do not, as a rule, try and say what will be done depending on the results, and I would submit that I think it would be in the interest of the designer at that stage, if they found a weakness, to correct it; or certainly from a regulatory standpoint if there was a weakness and not correct it, that we know about it at the OL stage.

MR. EBERSOLE: Mr. Chairman, as part of the TMI-2 findings, it was found and is now well known that there was something like a factor of 1 to 100 in the reliability of the aux feedwater systems. I think everybody now well knows that.

Does what you're proposing here really sort of extrapolate those sorts of studies into other systems in the plants to attempt to define what the spread of apparent reliability is in such systems as service water, component cooling, environmental controls, DC and AC power supplies, et cetera? Is it really sort of an extrapolation of the aux feedwater study?

MR. ERNST: That is sort of what it is.

MR. EBERSOLE: Do you have any guidelines for the applicants?

MR. ERNST: That's the next question, clearly is what is the acceptable reliability of these various systems, and the answer to that is no, at the present time, we do not.

As a matter of fact, I find it a little disturbing that we don't even for aux feedwater at this time.

MR. EBERSOLE: The applicants now will know what happened to the aux feedwater study. Perhaps they can step forward as volunteers and make proposals as what to do based on that as a starting point, although I haven't heard any such offers. I wonder why.

MR. ERNST: I can't answer that question.

MR. OKRENT: Well, let me ask the Staff what it is -- you know, is it really being done here -- is this something sort of cosmetic, or is it meaningful? If it's meaningful, how?

MR. ERNST: I wouldn't want to term it as cosmetic. I think over the years we will get a better handle on what kind of goals one would want to shoot for from the standpoint of reliability of other systems. I think it is a long-term goal, at least in the Task Action Plan, to look at reliability engineering, and clearly if you're going to look at reliability engineering, you've got to look at it from the standpoint of what kinds of goals are you shooting for in the standards.

The fact that we don't have the answer today, I don't think should preclude doing the kinds of studies that are recommended, but not superficially, but hopefully from the standpoint that we will be in better shape to decide what is reasonable and what is not reasonable from the standpoint of the subcommittee at a later time.

MR. OKRENT: I can't see how what you are proposing can have an effect on these plants, at least from what the Staff is asking, except at some later point after it's designed and built, and then you decide on some kind of a backfit.

In other words, I don't see any front-fit criteria or recommendations or whatever it is.

Now it's my impression that a variety of groups have said the single failure criterion as the Staff now employs it is not necessarily universally suitable or sufficient. I think even some members of the Staff say that.

How do you feed that into what you asked to be done in the reliability area, if at all?

MR. ERNST: Well, I think extrapolating from the single failure criterion is indeed getting into the reliability or risk arena, and that may -- well, I don't think Staff has a position regarding single failure or other kinds of deterministic ways of looking at things.

I'm just suggesting that maybe one way to find out about any weaknesses in the single application of single failure criteria is the risk or reliability overlap, and unless one does this, hopefully somewhat systematically on a number of plants, considering the uncertainties and things of that nature, it's difficult to make good regulatory decisions.

I don't think, however, that should preclude one from starting the analysis or starting evaluating systems to see what

does lie out there, and I think that's what this is aimed at, not the decision criteria and what you will do, but at least establish an analytical base for systems analysis to find out what lies out there as to forward fit or front-fit versus back-fit.

The requirements would be added. The only requirement on the CP stage would be the requirement to commit to do this program.

MR. OKRENT: To do the analysis?

MR. ERNST: To do the analysis and commit as to how it would be done, the timeframe it would be done, setting forth some objectives of the program, and this would be looked at by the Staff.

MR. PURPLE: If I may add one further element of what's asked for here, it says the program to ensure that the results of such studies are factored into the final designs means to me that we are speaking there more of a forward fit than a backfit.

MR. ERNST: Yeah, I was just going to work myself into that. I was just going to work myself into that.

MR. PURPLE: I'm sorry.

MR. ERNST: The words are there, and I think the construction permit licensees would certainly be more interested in reasonable front-fits than subsequent backfits. I think they have to realize that we would be serious about this program and should look at it from the standpoint of improving systems as

they are designed, rather than later.

MR. EBERSOLE: Well, in that context, has industry come forth with any voluntary proposals, or are they simply standing around waiting for you to create a basis to extrapolate? For instance, the aux feedwater studies. There are many such systems that could stand such studies, at least a dozen I can think of.

MR. ERNST: Yeah, there are a number of risk studies underway which clearly underlying that has to be reliability analyses of these systems.

You know, one could just say, "Do a risk study for every plant." My personal feeling is that the less sanguine -- that may give you some warm feelings, I don't know, but I would, I think, much rather be able to compare it against some kind of reasonably established methodology, so I know when I'm comparing one things, and I haven't got an orange versus an apple in comparing things.

We just feel like we want to do something constructive in this area. Asking for full-blown risk studies without establishing the ground rules for such studies, we think, might be not as constructive a use of manpower as maybe taking the first few steps ala this program, and maybe if methodology develops usefully in the next few years, and resources are available, converting this slightly smaller step into a more meaningful larger step, perhaps at some later time. At least

have the flexibility of doing that, if things develop properly.

MR. OKRENT: Let's see. It's over five years since the final version of WASH 1400 was published, and it's getting on two years since TMI occurred.

Apparently as of now, the Staff does not have either a general philosophy or specific approaches to individual systems of the plant with regard to what it thinks either is adequate or what it thinks should be modified, let's say, in future plants, aside from the existing single failure criterion, the standard review plan, whatever.

Correct me if I'm wrong, but at least I haven't seen written down something that goes beyond what was written down at TMI, except there are specific changes in practice that you have asked to be made, for example, as in the feedwater systems, you are not accepting certain things that you did not accept, perhaps unknowingly or whatever.

What is the procedure by which the Staff will decide what it's going to think should be done and when will that procedure come to fruition, in your opinion? And is there a plan in effect which will get one to fruition?

MR. ERNST: Let me say -- let me rephrase the question and make sure I understand it. Is the question really a better quantification or clarification of the backfit policy? When do you make changes?

MR. OKRENT: No. I'm asking about design requirements

for reactors from the reliability point of view.

In other words, that the various systems be reliable enough for the function they serve.

MR. ERNST: Okay I think the answer to that is yes, we have a plan. Whether it's extremely formalized and things like that may be certainly debatable. The action plan calls for a reliability engineering program that so far is to a large extent words, from the standpoint of the fact that you really have to have some reliability goals before you can establish programs and require licensees to test and design and maintain and so forth to a given reliability.

But I find it hard to separate reliability from risk, and risk from safety goals. So I think that there are -- I'm not saying we should not do any of these other things because of lack of a safety goal, but I tnink the three are tied pretty much hand in hand, and I think we are working in all of these areas, and I don't think it's dormant.

MR. OKRENT: No, I didn't say it was dormant, but I asked if there was in fact some kind of a well-defined program that would develop what you would call meaningful guidance in the design area, that presumably is the reason for your requesting this study by the NTCP owners.

In fact, it is unclear to me just how the Staff expects to arrive at these requirements. I have heard people say, well, IREP is going to give us insight and NREP will give

us more insight, and I have to agree that by studying these plants you will find first how they differ, and how in an incomplete sense, how some seem to be better than others for certain systems.

I say incomplete because I think IREP and NREP, as they are currently constituted, may in fact not only not review all aspects of the systems; they may even miss specific aspects of specific plants and come up with the wrong answer.

MR. ERNST: That's possible. Of course, NREP has not been formulated yet. We have IEEE or NRC or AIF --

MR. OKRENT: I'm just extrapolating from what I see in IREP. But I would have thought it possible the Staff would have some program specifically aimed at trying to develop -- change the criteria if it thinks changes are needed, and I guess I don't see it, and I cannot, therefore, quite see just what is the meaning of this particular requirement on the NTCP owners, in what way -- whether it's good or bad.

In other words, I can take two different points of view and not argue for either one. I could say, well, gee, there is certainly a thinking that the --- there could be improvements in design over the past. Why aren't you requiring these? Why haven't you designed these? Hasn't there been enough time?

Or I could take the point of view, well, we have a lot of plants running, and maybe their safety record is okay. Why are you requiring any changes? You know, there are other

positions, too, but I don't see now --

MR. ERNST: Let me see if this is the problem that you are addressing. The Staff is presently proposing or will propose shortly that we go back and take a look at reliability and risk assessments that have been done, specifically some sort of a regulatory sense, and identify the weak points in design and come up with some -- call them requirements or guidance or cautions or whatever, say when you are designing plants, here are the 12 things we have identified during the whatever to pay attention to.

Or one could say, based on our review of aux systems, aux feed systems, we think that Y reliability should be the minimum that you design for. You should demonstrate that you meet this.

Is this the kind of thing that --

MR. OKRENT: Could be, but --

MR. ERNST: I think Staff shares the concern, I know I do, that at some point you have to convert these reliability studies into some kind of regulatory requirement, whether it be prescriptive in nature or whether it be goal-oriented in nature.

I guess we haven't really sorted out. I think there are some merits to reliability numbers which clearly have to relate to risk of that particular sequence and things of that nature, but I think we are thinking about how do you take this experience and turn it into regulatory requirements.

Again I don't know if I have answered your concern or even addressed it.

MR. EBERSOLE: Dave, I thought the studies on aux feedwater and their findings would have provided a clear insight as to what could have been done next, and not this nebulous matter of just making generalized requirements on your reliability studies.

If we take all these plants and look at them now the way they are built, it's something like between eight to a dozen systems in the plant that contribute in various ways to, for instance, the rejection of shutdown heat out of the plant, for any kind of shutdown, short of a LOCA.

These systems make partial contributions to what should be another function, which is a dedicated heat shutdown removal complex, for that purpose, not to be used every day or every month, but when systems come apart.

I certainly would have thought that the studies on aux feedwater would have proceeded almost routinely to some of the studies on component cooling, of which we have a very modest study already inside ACRS, which will proceed in time to service water, environmental controls, et cetera, et cetera, on up to the eight to 12 systems, which I again emphasize each only furnishes some partial contribution to what should be a unified function.

Either the Staff or the industry should come forward with something less nebulous and we appear to be wandering around,

and until that occurs, I don't see anything but a stalemate.

MR. ERNST: I think this is a first step towards going down that road that you are talking about.

MR. SCHWENZER: Well, I guess the Staff considered this to be a worthwhile step to ask for these simple fact studies. It may not be everything that you'd like to have, but it seemed -- in fact, if my memory serves me correctly, I think this was an item that was suggested that we look harder at by the ACRS. I don't believe originally in the early look at NUREG -- what became NUREG 0178 -- 718 that even that much was a requirement. I believe that was added after our early meetings with the committee -- subcommittee.

MR. ERNST: I hope my comments haven't indicated I didn't think it was a worthwhile study. That's right.

MR. SCHWENZER: I was just trying to put a context. I thought this was something -- a step along the way that we ought to consider.

MR. ERNST: I think we are agreeing with you.

MR. SCHWENZER: Now there may be something better as we go along, and I think the -- this little insert page that you find in the thing tries to reflect that there may be a better approach. We are not trying to freeze it on the studies. If something better comes along, the Staff and the industry has a better approach before this work would be scheduled, we are certainly open to looking at that.

MR. OKRENT: Well, I suggest that we take a break, and we may or may not try to resume this before we go on to the next topic.

10 minutes or so.

(Recess.)

MR. OKRENT: The meeting will resume.

We were talking to Mr. Ernst. I wonder if I could ask him whether for the NTCP plants, the Staff envisages special approach by the utilities to what are now called unresolved safety issues for the eight items on the Staff's list. You know what I mean, in the TAPs A-1 through 40, whatever is the number these days.

MR. PURPLE: I'm aware of no plans to take these NTCPs on that subject in any special ways. In other words, the USIs are coming to fruition, and they are being applied to all plants as they get resolved, but there is no plans I'm aware of to devote some kind of unique attention to these set of CPs for those subjects.

MR. OKRENT: Now in auxiliary feedwater systems, for some reason the Staff, after Three Mile Island, looked at the existing plants and decided that certain minimum changes should be made on a short-term basis.

What is the rationale for having looked at those systems, but not at other systems?

MR. EBERSOLE: I could be cynical.

(Laughter.)

MR. OKRENT: Well, I don't want to be cynical. I may be getting skeptical, but I'm too young to be cynical.

(Laughter.)

MR. PURPLE: Well, I think it was -- probably it was the recognition of the major contribution that the aux feedwater system in the TMIX itself had been causing the problem that led us to look at that system hard, and I know of no other reason for that.

MR. EBERSOLE: It's like the death at the intersection before you put in a traffic light.

MR. PURPLE: I'm sorry?

MR. EBERSOLE: It's like the death at the intersection before you put in a traffic light.

MR. PURPLE: Yes.

MR. OKRENT: Let me ask this question: Is the intent of the reliability studies that you have requested that for PWR in the NTCP group, the reliability of the auxiliary feedwater system would be as good as the best of the systems you looked at to the extent one is able to assess this, or just that they compare it with these?

MR. ERNST: The requirement is for a comparison. What you do with the results that come in, the Staff has no standard at the present time, but it would strike me that it would be a very desirable thing to develop.

MR. OKRENT: Why should it not be as good as the best of those that you looked at? Could you tell me?

MR. ERNST: Well, I guess it's difficult to have everything as good as the best, but other than that, it's not exactly a proper answer.

MR. EBERSOLE: I don't think the best came about because it was difficult. It rather just happened, and the others happened to be poorer because of random rather than organized process.

MR. ERNST: I think that's probably true.

MR. EBERSOLE: And so what needs to be done is to organize those processes, rather than permit randomness to continue.

MR. ERNST: And this gets back, I think, to the before-break conversation, is there some useful regulatory requirements that one could glean from those studies? And I submit that that should be done.

All I'm saying is that it has not been done in an organized way today so that we could set such standards for the NTCP at this time.

MR. OKRENT: Okay. Well, I think we will probably come back to this topic again. Why don't we go on to the next one.

MR. PURPLE: The next topic would be degraded core rulemaking. Jim Meyer will discuss the alternatives we have been considering.

MR. MEYER: My name is Jim Meyer of the NRC Staff, and what I would like to do this morning is to run through the four options that are presently being considered in the area of degraded core rulemaking, and the implications of that rulemaking, and the requirements derived from that rulemaking on the near-term CPs and the manufacturing licenses.

On page C.4 of the Draft NUREG 0718, that's dated January 5th, you will see that the requirements under 2.B.8, rulemaking proceedings on degraded core accidents, that the original recommendation in the draft for comment had been removed, and the statement "alternatives under consideration" has been replaced.

What I would like to do then is to discuss briefly those four alternatives under consideration, and I would like to do it by starting with the option that has least impact on the construction permit and manufacturing license at this stage, and then move through them in order of increasing impact on the CPs and MLs.

The first option is basically to wait for the conclusion of the rulemaking before any action is taken in this area. Another way of stating it is to back off completely from the language used in the original draft for comment, NUREG 718, wait for rulemaking to be completed.

In this alternative, the applicant runs the considerable risk of major backfit as may be required by the

findings of the degraded core rulemaking.

This would be the same risk of backfit that the present operating reactors are going to be faced with, depending on the findings of the degraded core cooling rulemaking.

In summary, this option says that there will be no consideration of degraded core or coremelt in design or design modifications at this time, and as I mentioned, the licensees thereby run the risk of having to face the major backfit problems that perhaps could have been alleviated to some extent by taking on some of these questions at this time.

If there are no questions on that option, I will move to the second option.

The second option is a slight deviation from the first option, and for the most part it also is waiting for the rulemaking.

However, there are some actions that can take place at the CP and ML timeframe that I will go over very briefly.

Specifically we would state that this option would clarify the language in NUREG 718, that is the draft that was issued for comment, to say the Commission did not intend to require new features that would involve significant redesign and re-review by the NRC.

This implies minimal impact on the risk reduction mitigation feature considerations on the CP-ML licensing process in the schedule.

Minimal impact means the following:

That based on our knowledge of the appropriateness of mitigating features at this time, that certain requirements would be made of the facilities that we feel would be straightforward and would have minimal impact on the proceeding with a construction permit.

Several examples of these types of requirements would be to harden vital equipment so as to withstand a hydrogen burn event, similar to the hardening considerations presently part of the Sequoyah hydrogen study.

Another example would be to provide adequate emergency power for igniters, should the igniters option be chosen at a later date for hydrogen control.

Another example is to provide at this time large dedicated containment penetrations, and by large I mean a range of two to three foot diameter penetrations that would greatly aid in the orderly implementation of a filtered vented containment system, for example, if that was deemed appropriate at a later date.

These are examples of the type of requirements that would be imposed at this time. The emphasis, though, on this option is that these would not be major requirements in terms of impact on schedule costs or normal proceedings with the construction permit hearings.

Again, the licensee still runs the considerable risk of backfit, depending on the outcome of the degraded core rule-

making.

If there are no questions on option two, I will move to option three.

Option three is the original recommendation that was in the draft for comment, NUREG 718. It's a rather lengthy paragraph that I think you have seen before. Basically it requires two things:

One was that actions should be taken so as to not preclude or foreclose the modification of facilities which might be required by the rulemaking.

In addition, there was a requirement for the licensees to perform the study of mitigation features. This is different than option two in a number of respects:

First of all, it would probably require more significant near-term modifications to the design than option two.

Secondly, the emphasis in this option is on doing certain things that would not preclude or foreclose further modification, and I can give some examples of that in a second.

As Mr. Purple mentioned earlier this morning, one of the problems with this option is that it is open-ended, and it does not give specific guidance to Hearing Boards from which they can proceed in an orderly fashion with the hearings.

The more significant actions that would result in not precluding or foreclosing modifications would be such things as providing space under the vessel and the reactor cavity that

could possibly accommodate a core retention system. Space requirements, at least for certain of the systems under consideration, we feel can be estimated based on our Zion and Indian Point assessment of core retention systems.

Another option in this category would be a strengthening of the containment structure, depending on the status of the construction for these various plants. A very serious look would be taken at what now can be done to strengthen the containment structure so that the failure pressure could be considerably increased.

Another item would be the requirement for dedicated land areas near a facility to accommodate a large filtered, vented containment system.

These land areas were -- we feel again could be reasonably estimated from the work that we have performed for the Zion and Indian Point studies.

And another design consideration of this stage would be to make sure that the design could accommodate possible inerting either on a normal operational basis or inerting under emergency conditions.

The fourth option, and the one that has most impact on construction permit and manufacturing license stage, is one that closely parallels the present Zion and Indian Point study and mitigation features that is drawing to a conclusion this winter.

It would mean that there would be considerable immediate activity to address the questions of the appropriateness of mitigation features for these several plants, and to in fact take specific actions to see to it that the designs were moving on a schedule consistent with construction of the plant.

It would be anticipated that proceeding with this option four, that the plants would then be grandfathered in the sense of not having to meet the requirements of the degraded core rulemaking.

The program would be in two parts. The first part I have titled "Mitigation Strategies," and that first part would in turn be in two subparts. First would be immediate actions that would take place on a two to six-month timeframe.

This would include some detailed analyses of containment failure pressures, modes, and related analyses.

The second would be some specific design modifications that we feel hold out for considerable benefit and if in fact mitigation features are required in the future, such things that I mentioned before -- for example, of requiring penetrations, inerting capabilities, or the capability for installing hydrogen igniters, hardening of vital equipment, and such things as long-term containment cooling capabilities.

On a 12-month or a nine to 12-month timeframe, the utility would also be required to do a conceptual design study of mitigation features, and if you would like me to review the

eight mitigation features that are part of the package you have received, I would be glad to do so at this time, but it would be a utility study along the lines of what we have requested of the Zion and Indian Point utilities. That study being submitted to us, we hope, about the middle of February.

MR. SCHWENZER: Excuse me, Jim. I don't believe that was in the package, so if you want to get it in the package, why don't you throw the slide up.

MR. MEYER: If you would like to discuss the various features --

MR. SCHWENZER: Why don't you go ahead.

(Slide.)

MR. MEYER: I'll read these. Do we have a pointer?

Let me step back.

What we have done here is to put together what we feel are pretty much a complete set of potential mitigation features that would be worthy of study and consideration for CPs and MLs.

You can approach this list in different ways, and it could be reduced to several of the categories, but I think for talking purposes, it is an appropriate way to present the mitigation features.

The first one is energy removal for containment heat removal. Here we are talking about both active and passive containment heat removal, over and above what is presently

available.

Passive heat removal in particular looks very attractive in the long term. That is for long-term accident situations. We find this out from the Zion and Indian Point study.

The second mitigation feature is energy removal through containment atmosphere mass removal. The capability of removing the energy that you generated and preventing containment failure by venting that energy through a filtering system, as opposed to these other options, to have controlled small radiological releases.

The other options that have no release to the atmosphere over and above that which would take place from the normal leakage that would come from the containment.

The third option is energy dilution through increased containment volume. This is a subset of a filtered vent where instead of venting to the atmosphere, you would vent to a large volume leaktight structure along the lines of what some of the CANDO reactors have, I think, at the Bruce Station in Canada, where they vent to a vacuum building.

The fourth option is energy release suppression to suppression of burning combustibles.

And the fifth is closely related, which is the energy release management through a controlled burning of hydrogen and other combustibles, both these being approaches for hydrogen control.

The first would be such examples as inerting, water

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mist and halons, and the second hydrogen strategy would be the type of ignition systems, for example, being considered on the Sequoyah plant.

The sixth is energy release control and core mass management through core retention devices, such as core catchers, ladles, cavity flooding, both options of active and passive cooling.

Here you not only have the no radiation release to the atmosphere, but you have considerably reduced impact to the liquid pathway.

The seventh is here more for completeness, but it is the kinetic energy dissipation of steam explosition generated missiles through missile shields.

Again, if you have missile shields that prevent containment failure, you do not have any radiological release to the atmosphere.

And the eighth item, which may be appropriate at the CP stage and perhaps not for operating reactors, would be energy absorption enhancement through the strengthening of the containment structures.

T.3

The study that would be required of the utilities would be to perform conceptual designs using this list as a guideline, and hopefully in the nine to 12-month timeframe to develop the type of conceptual designs that we anticipate will be forthcoming from the Zion and Indian Point study, which is due

next month.

These design features would, of course, have to be consistent with the plant designs and the plant modifications that would be done on a more immediate basis.

The second part to the option -- the first part was the mitigation strategies. The second part is a more formal risk analysis.

This we would anticipate taking approximately 18 months, and the purpose for this risk analysis would be to determine which of these features is best in terms of risk reduction.

An important ingredient in this risk analysis would be the consideration of competing risks that are taking place, the most important of which are the competing risks that are introduced by the actual installatio of mitigating features.

Like I said earlier, this option would be closely paralleling the Zion and Indian Point activity, with hopefully the same results coming out of it. That is specific requirements at this stage is felt necessary for a specific mitigation feature.

Those are the four options.

I will be glad to answer any questions regarding them.

MR. MATHIS: Jim, on option three, have you taken a look at any of these particular designs or the plants or sites, to see what impact these might have on the --- could they be accommodated at this stage of the game, such as your space under the unit, the land area for filtered, vented containment, and so

forth?

MR. MEYER: That would be a suggestion for consideration, and that comes out of our experience with the Zion-Indian Point study, whereas you are probably aware, there is very, very little space available in the reactor cavity to accommodate the type of core retention system, for example, that is now part of the Offshore Power Floating Nuclear Plant design, and the idea here would be to have a significant amount of space available. That space could always be filled in later, as it was to a certain extent on the FFTF reactor. But it would be available to provide considerably more flexibility, in proceeding with the core retention design, if that is deemed necessary at a later date.

MR. OKRENT: Let's see if I understand what your alternative four is. After issuance of a construction permit, or at least not before the issuance, the utility would have to do conceptual design studies on features or systems related to the eight items you have on the viewgraph, and it would also have to assist the risk reduction potential.

MR. MEYER: The 12-month timeframe that I indicated for the actual conceptual design study is intended to be completed before issuing of the construction permit.

MR. OKRENT: Before issuing? Now what would have to be done before the construction permit then specifically?

MR. MEYER: The first two items that I talked about --

the first one was immediate actions. Again these are plant-specific, and there would have to be a determination on a plant-by-plant basis which were appropriate actions to take, but there would be such things as making sure that there were adequate containment penetrations, adequate emergency power for hydrogen igniters, hardening and placing of vital equipment to withstand hydrogen burn environment, items like this.

This we feel could be accomplished in a two to six-month timeframe. Paralleling that, that is in parallel with that immediate action part of the program, would be the utilities' study of these seven mitigation features directed at very practical, specific questions regarding the capability of these features to work as they are conceived to work, and the making sure that whatever features are studied, are consistent with the plant design as it is presently progressing. And these two components -- at least our first thinking -- these two components would take place prior to the issuing of a construction permit.

MR. OKRENT: Suppose the utility had done this, and managed to do this in a 12-month period, you have just indicated, would you have to review this in some way and come up with some kind of an evaluation of this prior to the construction permit?

MR. MEYER: Yes, that would be part of it. I would envision that review to be similar to the review that we were anticipating to do for the Zion and Indian Point study.

MR. OKRENT: That's not part of the same 12-month

period?

MR. MEYER: That would be included in it. Again I am using as a guide what the Zion and Indian Point utilities have been able to do over the timeframe of starting in December and now being able to -- we pretty much have that study complete in about a year's timeframe.

MR. OKRENT: Well, but I'm just trying to see what this alternative comprises at the moment, at least so I have a picture in my mind of what it is you are saying. In the 12-month period they would be asked to do these studies, and I think you said -- but not the estimate of the risk reduction potential. That was a longer thing?

MR. MEYER: Well, as long as that we are assured that then proceeding with the granting of a construction permit and the construction that would ensue in six month to possibly a year interim, as long as we would be assured that that would not get in the way of actually installing these mitigation features, then the risk study which will be the final yardstick for determining which of these seven, if any at all, are going to be required, that study can take a somewhat longer time; in the range, as I said, of 18 months.

MR. OKRENT: Now let's say that they had done these studies. How would the judgment be made as to whether or not any or many of these features should or should not be implemented?

MR. MEYER: Well, let's assume that the risk

analysis draws the conclusion for one of these reactors that hydrogen burn is the major threat to the containment integrity, and that steam overpressurization is a relatively minor aspect of the total pressurization of the containment.

Then we would be in a position to -- within the limitations of a risk analysis -- to determine the benefit from installing hydrogen protective measures, and would probably see a considerably greater risk reduction in that then, for example, the installation of a filtered, vented containment system where the primary concern there would be for accommodating a slow overpressurization from the steam generators.

MR. OKRENT: Now you used the term "within the limitations of a risk study." That's a good term to use, but what are those limitations, do you think, and how will they impact on your ability to make assessments and judgments?

MR. MEYER: That is a key question that we are presently taking on for the Zion and Indian Point study, and one that we are going to be forced to answer as part of the Staff report on the Zion-Indian Point study.

My present opinion would be that it is a -- it is an effective tool-methodology for translating engineering judgment that would come out of something that said hydrogen is a bad problem, and the focus should be on hydrogen control, translating that into, for example, consequence reduction that the technical community is very familiar with.

In that sense, we would be able to have a handle on, for example, reduction in latent cancers or early fatalities that would result from the proper operation of the specific mitigation feature or a combination of mitigation features. But I certainly would note at this time, simply quote the one consequence number, assuming containment failure, and the other consequence number, assuming, for example, the filtered, vented release of a small amount of radioactivity.

But it is a very difficult question.

MR. OKRENT: I mean, for example, you've got one item there. It's No. 3. It says energy dilution through increased containment volume venting to large volume leaktight structure. No radiological release to atmosphere. That's what it says.

Of course, there are some assumptions that have gone into that statement of no radiological release to atmosphere, obviously. You have ruled out those accidents where in fact your first containment has lost its integrity for one reason or another, and so there will be a certain set of scenarios where this may not be effective or partly effective.

MR. MEYER: That's true.

MR. OKRENT: It depends on the design and so forth, and one person might do a risk study that says yes, this is essentially the case, and somebody else might do one -- or a person might do it two years later, coming up with a different conclusion.

Now I'm trying to again get the decision-making process and to see whether it is practical and how you envisage doing it, and so forth. I'm not quite sure at the moment.

MR. MEYER: Well, like I said, it's a very difficult problem. We have a certain state of technology and rulemaking now is proceeding with certain research programs, and other programs in place specifically to address what we feel are the weaknesses in the state of technology, and one of those is how to answer that final question of the net risk reduction from any one of these or a combination of these mitigation features; in particular, taking into account the competing risk, whether that competing risk be an event that would be present, anyway; containment isolation; failure; or those competing risks that would be present because you installed the mitigation feature. And we are trying, because of the state of technology, to emphasize that particular aspect, what can go wrong if the system is installed aspeet, and the Zion-Indian Point evaluation that we are doing, and it would have to be an important element of this study.

But the option is assuming that something is to be done now as opposed to the other options, which to a certain degree say we are going to wait for the findings of the degraded core rulemaking.

MR. MEYER: Are there legal opinions among the Staff on this one that this is a tenable approach with regard to the

legal process?

MR. MEYER: The question of grandfathering?

MR. OKRENT: In other words, if the Commission adopted something like what you would call alternative 4, that the Hearing Boards could deal with it and so forth and so on.

MR. REIS: Well, it probably is possible, but we haven't looked at it in depth. It is essentially a policy question, whether you want to do it that way, and have the Board spend that much time in order to come to a proper conclusion.

MR. OKRENT: Would the Board have to go through all of this in detail to see whether the studies had --

MR. REIS: Well, it depends upon the wording of the rule that the Board would be bound by, and how much freedom or how much control you put in the rule on the Boards and how far they should go.

MR. OKRENT: Somebody would have to look at this. I assume the Commission would want this all done without it being evaluated, so it would be the Staff and the Board, or the Staff or the Board, I assume.

MR. REIS: That's right, and it's a policy decision for the Commission to make as to where to put it.

MR. OKRENT: I guess at the beginning I said I was interested in hearing the pros and cons of various alternatives. This seems like a good simple area in which to talk about pros

and cons. Why don't we hear the pros and cons of 1, 2, 3, and 4? Maybe you can say what they are again, briefly, and tell us what they are.

MR. MEYER: I think you can divide the pros and cons into tenable pros and cons which the Staff can appropriately address at this time. I guess there are policy pros and cons and legal pros and cons. I won't dwell on those, but the first option was to essentially remove the 2.B.8 requirement from the NUREG 718, have no requirement at this time regarding the accommodation of degraded coremelt accidents.

The pluses, the orderly proceeding, at least from that standpoint, of the construction permit and manufacturing license proceedings. The negatives -- the technical negatives would be that we are not recognizing that we have learned something over the past year and a half regarding mitigation features. And that that knowledge should be put to good use at an early stage in the construction of these plants.

The other negative that I mentioned regarding this option is that the full burden of meeting the requirements of the rulemaking in terms of backfit would be imposed on these facilities, as well as the operating reactors. That, of course, will have to be backfit.

The second option was a slight variation on the first option, but to require certain -- well, minimum considerations of degraded core coremelt and the design or design

modifications at this time.

These would be the type of modifications and studies that again, based on our experience with Zion-Indian Point, we feel are in pretty good shape technically, and would have minimal impact on the construction permit itself, and I gave several examples.

The disadvantages would be that again technically we feel that more requirements can be imposed at this time that would give us greater flexibility later on, but those additional requirements would be more disruptive in terms of construction permit hearings and construction itself.

The advantage, of course, is that the construction pretty much could proceed on a normal schedule, and the other disadvantage is that again there is the risk of major backfit coming out of the rulemaking requirement.

The third option, I think the biggest disadvantage of the third option -- and the third option was exactly the wording of the draft for comment of NUREG 718 -- was that it was too vague and it would not give enough guidance to the Hearing Board.

Such things as guaranteeing that the construction permit stage would not preclude or foreclose the modification of facilities which might be required at a later date. Technically if we are going to talk now about major modifications, that would not preclude eventual installation, but it's very difficult to be able to make those decisions without a thorough analysis of

the risks from the individual plants, and the status of construction of the plants, and the capability or the benefits of one feature versus another.

The possible advantage is, that if it is done right, then all of the major obstacles have been taken care of for meeting the rulemaking requirements that come up at a later date.

The fourth option has the disadvantage of a potential delay in construction of these power plants. However -- and it's depending on a lot of policy and legal questions --- there is the potential for the grandfathering which would exclude them, these plants, from meeting the requirements of the findings of the rulemaking.

MR. OKRENT: Now let's see if I can understand what you mean. What is it that they would be grandfathered from? They would presumably have done conceptual design studies on your eight features here, and they would have examined the risk reduction potential of these, and in some way then presumably the Regulatory Staff would arrive at a decision then that 1, 2 and 5 or some other combination thereof should be pursued.

Now, presumably, then, if that recipe were followed, the Commission could say that having been done, they would be grandfathered from the future rule, whatever it may be. Is that what you are saying?

MR. MEYER: No, that's what I understand the grandfathering to mean. I don't know if there is any further comment

on that from representatives here from Legal, but as I understand it, and I am certainly not the one to ultimately ask the question of, that would be basically --- what you said was basically what the grandfathering means.

MR. OKRENT: Is that --

MR. REIS: It depends upon how the Commission writes the regulation. Essentially what the approach talked about, I think, was an ad hoc approach. In other words, those plants before the rule takes place would be considered individually on an ad hoc basis, and they would be excluded from the future rule that would take place, and that depends again on how the Commission writes the regulation.

MR. OKRENT: I'm lost now. You said they'd be considered on an ad hoc basis. Is it this ad hoc basis or some other?

MR. REIS: This ad hoc basis that he just described, it is his basis, whatever basis we put in the regulation, in the rule, is the basis upon which they would be considered.

MR. OKRENT: And they could be legally grandfathered then this way, so then the rule later would not apply?

MR. REIS: If the Commission so wrote the regulation, yes.

MR. OKRENT: Well, I know that Congress can write a law one year, and the next year can write a law that repeals the law. I didn't know if the Commission could also do that. I

suppose.

MR. REIS: There could be some problems from the point of view -- I don't think there are.

MR. OKRENT: In other words, can this Commission grandfather something from a future Commission?

MR. REIS: I can't answer that question off the top of my head.

MR. OKRENT: Can I ask a technical question? On No. 8, it says energy absorption enhancement through the strengthening of the containment structures. Suppose I consider one of the low pressure types which I guess is either an ice condenser or a Mark III. What does that mean to you, that No. 8? What is it you think they are going to do in the study, and what would they do in revisions and design of construction under 8?

MR. MEYER: Well, I really can't speak to that in terms of any specifics. I know that the Staff is looking at that question. The structural engineering branch is addressing the question of what modifications to the containment structure would provide for more margin to the failure, and I could guess as to some of the approaches, but I don't have a real good handle on the specifics at this time.

MR. OKRENT: But presumably this is a decision that would be made prior to construction?

MR. MEYER: It would be a decision that would have to be made consistent with the construction; that is, if it was to

add further thickness and rebar and whatever to the containment walls, the construction would not have proceeded to that point, so that that could be accommodated.

MR. SCHWENZER: There is a couple of possibilities. One being a relative increase in strength by a larger volume. For example, going out vertically, would be one possible way.

Another way is thickening or strengthening the materials of construction. For example, a thicker steel wall.

Those are things that probably go towards less perturbation to the design work that's already been accomplished on these plants, or anything else in terms of trying to get the larger or major structural things, probably would be major redesign sorts of efforts.

I guess a lot would be dependent on what would be proposed.

MR. MEYER: Yes. Again it's down here as an addition to the more standard list for the operating reactors, because there is the potential at this stage for considering the appropriateness of that approach. It may very well be that that approach is considered totally impractical.

T.4 MR. OKRENT: Now I agree that there is a potential, but before you start construction, it's not clear to me how much of it remains after you have ---

MR. EBERSOLE: Is item 8 oriented specifically and only to hydrogen explosion problems?

MR. MEYER: It would be certainly a contribution to be reducing risk to help hydrogen from burns or possible explosions, but it would also be a big help in terms of steam generated containment loadings, too, for some of the small volume containments.

MR. EBERSOLE: Steam explosions.

MR. MEYER: No, this would be just a slow steam pressurization. It probably wouldn't be much -- well, it would contribute somewhat to the question of the steam explosion.

MR. EBERSOLE: In item 1, where you talk about passive heat removal, are you talking about passive elements interior to the containment, with active elements exterior to get the heat out, like a flooded exterior wall?

MR. MEYER: By passive I mean no -- you can walk away with no AC, and [illegible] that type of passive to one where you could, by various means, remove heat from outside the containment by interdiction.

MR. EBERSOLE: Without active elements inside?

MR. MEYER: Without active elements inside. The active component would have to be from the outside.

MR. EBERSOLE: Like flooding or spray?

MR. MEYER: Right. Uh-huh.

MR. OKRENT: Is the Staff leaning towards one of these four, or away from all of them?

(Laughter.)

MR. MEYER: Well, assuming that this represents a relatively complete set of options, the Staff will be making a recommendation for one of them. I really think it would be inappropriate at this time to state where the Staff is heading on this issue.

MR. PURPLE: Inappropriate may not be the right word, just because I don't think we know yet where we lean. We are going to develop that through the rest of this week, and in fact we may not have a specific recommendation to the Commission. It may be a presentation like this with,however, a more precise answer to the question you just asked. When they ask it, as to where do we lean, and which one do we think is the best overall. But we really haven't reached that point as of yet, today. So it's not a matter of appropriateness, it's a matter that we don't have the answer.

MR. OKRENT: Does the Commission provide any policy guidance to help you in this beforehand, or do they wait until you come in with the alternatives?

MR. PURPLE: They haven't in this area yet. I anticipate and hope we will get some when we see them next Tuesday.

MR. OKRENT: Getting back to the area Mr. Ernst was discussing, are there pros and cons that we should hear about there? That was presented as a single alternative.

MR. ERNST: Well, I think, if I understand your

question, I think that the start of my impromptu discussion did pose several alternatives and some rationale of why we selected the one proposed. One alternative essentially to do nothing until you get a safety goal. I did not give pros and cons of that, if that's what you're looking for, but I did mention three possible alternatives.

MR. OKRENT: Want to tell us what the pros and cons are?

MR. ERNST: I guess the pros and cons of doing nothing, I think the record and people's observations of what should be done really doesn't support doing nothing. I think there is enough that has been done in reliability and risk assessment, enough understanding of the basic methodology, and to some extent the application that doing nothing in the Staff's view is just not appropriate. While one is striving for a safety goal, I think the real question of when and maybe even whether and to what extent a safety goal will address this problem, I think is still somewhat problematical. So I think that do nothing is not appropriate at all.

There have been some benefits from doing risk and reliability studies and there may well be some benefits -- the Staff thinks there are some benefits of continuing.

The other option that was discussed, I guess, was requiring a risk study prior to issuance of an OL, and I think there is a fair market for supporting that kind of a position.

The con of supporting that kind of position is sitting here today, we really haven't formulated a good priority system for which plant should be done first and we still have not developed a good reasonably standard methodology for performing the assessments.

So to make the decision today that you shall do a full-blown risk assessment before an OL issuance is probably premature. So I think the proposal is to allow that kind of flexibility, but not to mandate it at this time.

If two or three years from now, it looks like it's rational from a priority standpoint and the resource from availability standpoint and methodological standpoint and decision-making criteria setting standpoint to so mandate, then that could still be mandated and replace the requirement for more simplified reliability studies and specific systems.

MR. OKRENT: Okay. Jesse?

MR. EBERSOLE: In looking at these eight items here, in looking at them in the context of having eliminated the requirement to provide accommodation for core ladles, it seems to me all eight of these are interdependent. None of these lead to enhancing the chance that you will not have a severely degraded core. They all pertain to containment heat removal.

MR. MEYER: They are all mitigating features under the assumption --

MR. EBERSOLE: So I look at them really -- I can't

help but say, look, this means core retention devices are likely to be imposed, as well as these other things. None of these are a substitute for core ladles at all.

MR. MEYER: Maybe I'm confused by what you're saying. One of the options is a core ladle.

MR. EBERSOLE: Well, it's not a -- look at the other seven, they have nothing to do with improving the prospect of not having a molten core. They only pertain to the protection of the containment from physical effects other than meltdown -- meltthrough, you know, like explosion and heat removal.

MR. MEYER: Well, the same is true for the core ladle. All the core ladle does --

MR. EBERSOLE: It has a special function.

MR. MEYER: The special function assumes that you have vessel meltthrough where you have core on the floor.

MR. EBERSOLE: Right. None of these other things in here appear to be in the character of reducing the potential for that.

MR. OKRENT: They would fall in Mr. Ernst' bailiwick. Jesse, if you want to pursue this, I would suggest you try to see whether you have something more specific. In other words, is there a fourth alternative that you would put on Mr. Ernst' list. He gave three. They wouldn't be on Mr. Meyer's list. He's been asked to address the mitigative part. Mr. Ernst is addressing the preventive part, if I can categorize it that way.

MR. EBERSOLE: Well, what I'm saying, these are not alternatives. They still have the core --

MR. OKRENT: Yeah, but they're all in the mitigative area, and he's not trying to deal with the preventive area on this list, not at all.

MR. MEYER: That does not diminish the importance of prevention at all, it's just a new defense-in-depth consideration that came out of the TMI accident plan and other studies.

MR. OKRENT: Well, let's see. According to the agenda which is by no means a Commission rule --

(Laughter.)

-- we are supposed to get to the NTCP plant owners. Let me ask, are there any legal insights that the NRC Staff would care to share with us at this time beyond those we have heard?

MR. REIS: There would have to be specific platforms to give them. In the absence of whether this plan and what is going forward has been looked at by Legal. Yes, it has been looked at by Legal. We have had a man on the committee working with us continually developing this. We think that what is proposed here are matters that with the proper regulation of formulating and putting into effect -- I don't know what else you are asking for, from the point of view of legal insight.

MR. OKRENT: At the moment, it was a general question. I may become more specific later in the day. I just was --

MR. REIS: I'll be here.

MR. OKRENT: Well, should we begin hearing from representatives of NTCP, or shall we break for lunch?

MR. MATHIS: Let's break for lunch.

MR. OKRENT: Jesse?

MR. EBERSOLE: It doesn't matter to me.

MR. OKRENT: Is there a problem with anybody if we break for lunch at this time?

All right, let's reconvene in one hour.

(Whereupon, at 12:15 p.m., the meeting was recessed, to reconvene at 1:15 p.m., this same day.)

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AFTERNOON SESSION

(1:15 p.m.)

MR. OKRENT: The meeting will reconvene.

I am sorry I missed Mr. Schwencer this morning in looking down the outline of the Staff's presentation, so we will take that and then proceed to the NTCP representatives.

Mr. Schwencer?

MR. SCHWENCER: My name is Al Schwencer, Division of Licensing.

(Slide.)

As a result of the public comments received, we did make a revision of the NUREG-0718, a draft of which we made available this morning. A number of staff members in each of the NRR Divisions, including now the Office of Inspection and Enforcement, participated in evaluating the comments, and also looking internally and making a number of revisions.

The more important ones have already been discussed with you this morning -- or at least, in our view, the ones that seem to merit the most attention.

Just for the purposes of indicating that comments were received from two private individuals.

(Slide.)

The applicants are represented within this group. As a result of going through the comments, and also taking,

in effect, another look-see audit at the classifications: If you'll remember, the Staff earlier had categorized the requirements, or the "nonrequirements," into five categories, one through five.

We took another look at that. There were some editorial errors in the package, and some judgments that some of the items should be elevated; and in other cases, they should be relegated to a lower category.

For example, "quality assurance." We felt that is something that should normally be looked at at a CP stage, so we elevated that to a Category V.

(Slide.)

Just briefly, I would like to show these next two slides. They are on the handouts that are being passed around. What we have shown on the left column is the categorization that was made in a preliminary instance. CPs and then a "/" ML categorization, because the manufacturing license doesn't have some of the responsibilities in the regulatory review that a CP owner-applicant would have -- a number of the things, such as siting, did not apply to the manufacturing license applicant.

The second column are the reassigned categories. I have -- There are essentially two shades. Bob Purple touched on these earlier. What I would like to do is just show them briefly right now.

(Slide.)

Those that were reassigned to Category I in essence said that for the CP, the prerequisite to CP, these need not be addressed at this time. Those reassigned to Category II -- the "liquid pathway," for example, at the bottom on the MLs; that we changed it from one that would not be a requirement of the ML license, but certainly would be one as soon as a site had been located, and would need to be addressed.

In addition to making the recategorizations, we also found a need to revise the language -- expand, and in some cases add items. For example, II.F.3, some of the comments received from the public was that Reg Guide 1.97, it looks like it was going to be a long time before that got issued. In fact, Reg Guide 1.97 was issued December 24th. So we revised the language to bring that up to date.

In some cases, it was a simple case of adding a few words, adding a reference to 0625 in one area. In the siting rule which we have already discussed today, reflecting the fact that indeed that has slipped; it was called "October" earlier; it's now projected for April of this year.

Generally, what I would propose is that I have a number of the group that worked on the revision here today, and if you have specific questions on any of these changes we would endeavor to answer your questions. Otherwise, the

revised draft we will let speak for itself.

MR. OKRENT: Are there any specific questions from the subcommittee?

(No response.)

MR. SCHWENCER: We would intend to make these same slides available to the Commission in the Commission Briefing paper.

MR. OKRENT: Okay, I guess then at least for now we have completed the Staff's presentation?

MR. PURPLE: That's correct.

MR. OKRENT: So why don't we go on to the next step. I believe Mr. Walker has a presentation to make. Why don't we give him the floor.

MR. WALKER: I am D. Walker of Offshore Power Systems.

Today I would like to talk a little bit about where our license application stands, and just briefly review that, and then summarize our reactions to the proposed rule that was issued for public comment. We of course had no opportunity to see the Staff Revision before coming to this meeting, so it is difficult to be prepared to comment on that.

(Slide.)

MR. OKRENT: We are in the same position you are.

(Laughter.)

MR. WALKER: The floating nuclear plant review, the

status is summarized in this viewgraph. The NRC has issued an SER and three supplements, and they've been reviewed by the ACRS. The ACRS has issued interim letters on our application, including a letter on the core ladle. We have a need for at least one more SER supplement following staff review of the OPS responses to the TMI Action Plan.

OPS responses to the TMI Action Plan, NUREG-0660 were prepared last summer and submitted to the Staff last July. We also provided copies of those responses to the ACRS at that time. The only item that was not addressed in those responses was the degraded core rulemaking, and I will try to discuss more where we think we stand on the degraded core rulemaking later in this presentation.

With regard to our hearings, our hearings are completed except for TMI matters. All the existing contentions have been heard, and partial findings of fact have been filed with the Board by both OPS and the Staff. Except for the TMI matters, the FNP licensing process is essentially complete.

With respect to the notice of proposed rulemaking issued by the NRC for public comment, we reviewed that proposed rule and submitted comments on the rule. If the Subcommittee would like copies of those comments, we could provide those today.

MR. OKRENT: I believe we have copies of all the comments that were formally submitted during the time for such

comments.

MR. WALKER: Okay.

The proposed rule set forth three approaches, they being: using pre-TMI requirements augmented by the requirements of NUREG-0660 as modified for the NTCP/MLs by NUREG-0718; the second option was: take no further action pending the rulemaking action, their completion; and the third was: Resume licensing using the pre-TMI requirements augmented by NUREG-0718 and certain additional measures or commitments in selected areas that have been discussed extensively this morning.

We at Offshore Power Systems prefer Option 1. We do believe, however, that either Option 1 or Option 3 could be used for resumption of plant licensing in the near term with no sacrifice in ultimate plant safety. We believe that resumption of licensing reviews for the NTCP/ML applicants and their applications is long overdue, and we urge that a decision to proceed be reached in the near future.

As far as we were concerned, the recommendations in NUREG-0718 could be divided into five general categories. I have a viewgraph that summarizes them in the way that we looked at them.

(Slide.)

We feel that these categorized are: the plant-related design requirements; those associated with degraded

core conditions; administrative controls; operator training and actions; siting, evacuation, and emergency planning.

The last two of these categories, as I have indicated here, are not applicable to the manufacturing license since they are the responsibilities of the plant owner and operator.

As I mentioned, we submitted to NRC responses to the recommendations of NUREG-0718 in mid-July, which were related to the plant design and related to administrative controls.

These relatively detailed responses were contained in a 120-page document. We believe our proposed implementation of the recommendations of NUREG-0718 were both responsive and adequate -- although we have not received any comments from the Staff and, so far as we know, they have not reviewed our material.

Item II.B.8 of NUREG-0718 of course concerns the rulemaking on degraded core accidents. Our response to NUREG-0718 postponed response on this subject till an interim rule was in place. However, as the subcommittee is aware, we have already taken significant action with respect to degraded core conditions.

We have complied with the NRC Staff requirement to add a refractory ladle device beneath the reactor vessel. I would like to emphasize that the purpose of this device was to

delay the release of [illegible] to the basin in which the FNP floats, not to prevent it. The purpose of the delay was to provide sufficient time for interdictive actions to be taken.

In addition, there was a substantial evaluation performed regarding degraded core conditions as a result of questions posed by the ACRS in a letter of July 25th, 1979. Included were questions and responses on the subject of dealing with the buildup of hydrogen in a containment building following an accident, and also with the flexibility in the design to accommodate design changes for mitigating the effects of degraded core accidents. These OPS responses to the Staff and ACRS were submitted in mid-September of 1979, over a year ago, with the staff evaluation following some two months later.

(Slide.)

My next viewgraph deals with the requirements of NUREG-0718, and summarizes in a little different format the area of special attention where we think we stand with regard to them.

As I mentioned before, two of the areas are not applicable to the manufacturing license -- the emergency planning area, and the siting area.

I addressed our earlier actions with respect to degraded core conditions. More recently, of course, the NRC issued for comment a proposed rule on degraded core conditions which we feel is intimately tied to resolution of the overall

question of degraded core conditions for the near-term construction permit manufacturing license -- although the Staff has not discussed that interim rule situation today, as yet.

Generally we believe the proposed rule was too proscriptive, focusing on possible mitigating devices rather than setting forth criterion and guidelines which would be a basis for design.

However, we do support the approach of early adoption of an interim rule as a basis for proceeding with near-term licensing. We believe that such a rule should be supported by a Commission policy statement which indicates that the rule does provide the basis for proceeding with licensing until the final rule is issued.

We also noted in our comments that a near-term construction permit manufacturing license policy statement requires that applicants submit proposals for mitigating features prior to the issuance of a construction permit or manufacturing license.

In contrast, the proposed interim rule requires that the studies of possible mitigating features be submitted within six months from the effective date of the interim rule, or the date of docketing of the application for the operating license, whichever is later.

We particularly support the latter requirement. We support and strongly urge that the latter requirement be

adopted, rather than the near-term one. We feel that to do otherwise would result in a substantial additional delay with no commensurate increase in safety. Alternatively, applicants would be encouraged to engage in a hurried design effort providing little more than hardware for the sake of hardware.

We recognize that hydrogen is the degraded core issue with the most potential for design impact as a result of TMI-2, and also recognize the ACRS interest in this subject particularly for smaller containments.

We are participants with the other ice condenser plant owners in analytical studies to evaluate the need for and the effectiveness of hydrogen control measures. We do not propose to specify particular features for hydrogen control at the present time, but we would expect to add such features in the final design if such a requirement results from the degraded core rulemaking.

At present, ignitors for controlled hydrogen burn appear to be a feasible approach which offers substantial protection. We have also evaluated the pressure capability of various regions of the containment, and that evaluation was submitted in the September 1979 responses. We believe that design alternations to raise the pressure capability of our containment could be accomplished should such become a requirement.

With respect to proceeding with licensing, we believe

some hydrogen control criteria or requirements should be identified in the near term that are comparable to those under which current operating licenses are being issued. Certainly the Sequoyah decision provides a suitable basis for proceeding with licensing of ice condenser containments.

Regarding the reliability analysis, which was Special Concern No. 3 in the proposed NTCP rule, OPS believes reliability evaluations can and should be factored into the design process. As part of the final design, we have committed to perform reliability evaluations for systems important to safety.

The objectives of these analyses will be to establish overall systems reliability estimates, and to identify the principal contributors to the potential system failures. Particular attention will be paid to identification of operator errors, common modes, single failures, and test and maintenance outages which contribute significantly to systems failure probability.

The results of these analyses will form the basis for appropriate systems design modifications, if required. The systems reliability analyses are to be submitted to the NRC within two years after the issuance of the manufacturing license.

We recognize, however, that the application of these techniques to the decision-making process requires that a

safety goal be established and that a body of failure-rate data be accepted. We encourage the NRC to take the positive steps necessary to make reliability engineering a meaningful tool in plant design.

The last item here is the SRP Acceptance Criteria which was discussed in the NTCP/ML rule --

MR. OKRENT: Excuse me. Before you go on to that point, the last full sentence back you stated was something like: You urge that the NRC take steps to make reliability engineering a practical step and an actual step in plant design. Something like that.

MR. WALKER: Yes.

MR. OKRENT: Why do you think it is for the NRC to do that, and in your case not for your own company?

MR. WALKER: Well, Dr. Okrent, I think that's the responsibility of both parties. I think from our reliability analysis certainly we can identify the outliers that contribute significantly to risk, and take some action to correct those or make design modifications.

However, there comes a place where our decision and those of the NRC might be quite different. I guess I feel that we rapidly reach -- we would rapidly reach that kind of condition were we to estensively apply reliability analysis in the design process. I feel that some kind of unified approach or goal to making decisions in those areas where there

is question is needed by the NRC.

MR. OKRENT: If I may, I would like to explore this area some.

You are the designers, not the NRC. In principle, you know your plant better. Also, you have a feel for where there are constraints, and what it is that imposes the constraints, and so forth.

So my feeling is that in fact you are in a much better position than the NRC to do design which is influenced by reliability than for them to specify something that you should do.

Have you, up to this point, done what you would consider to be a reasonably thorough and systematic reliability analysis of your own design to judge to your own satisfaction that (a) there are no areas where in fact, without great changes in cost, you can improve the reliability of important systems, whether they be safety or not; or (b) with some increase in cost, you can achieve some potentially significant improvement in overall safety?

In other words, has that kind of analysis already been done by you, and you think you are optimum in this regard? Or is that something that remains to be done, in your opinion?

MR. WALKER: I have a mixed answer. We did kind of an overview mini-study on this plant, and we have identified some areas and made correction without great cost.

However, we have not done the systematic kind of evaluation which you just described.

MR. EBERSOLE: Did you find your aux feedwater system were adequately --

MR. WALKER: We did the aux feedwater system evaluation and found that we were at the higher end of the reliability scale with that system, compared with other plants, at least with that system.

MR. EBERSOLE: Did you do anything on the scram system?

MR. WALKER: We haven't done the scram system.

MR. OKRENT: Now would you view it as both reasonable and appropriate for the FNP not only to do what I'll call some kind of "systematic study" to assess your -- there may be weak points; but to do it in a time frame such that in fact weak points, significant weak points can be immediately corrected? That's two different things, and I want to make sure that we distinguish between the two -- between assessing the system as designed; or doing an assessment in a fashion where you can in fact make what you think are significant improvements.

MR. WALKER: Well, the answer to both questions is really "yes." We think an adequate reliability assessment of this system needs to be done. And we think that if we are going to do it, we ought to do it in time to impact the

design, if the assessments show that that is necessary.

We feel, however, that that can be part of our final design process and need not be done before the manufacturing license.

MR. OKRENT: Now do you have some words that you would propose to the NRC that they use that in fact would end up in your doing what you just said?

MR. WALKER: Well --

MR. OKRENT: Because I don't think that's what is in what I heard Mr. Ernst write, because that was a little bit more fuzzy to me. I didn't quite get the sense from his words that I think we just used.

MR. WALKER: Yes. I thought the words that we wrote in our response to the NUREG-0718 had that kind of a flavor to them on the reliability engineering question. I don't have that big document with me.

MR. OKRENT: We have a copy -- No, we don't have the big one. We have only the comments that were sent specifically to this proposed NTCP thing.

MR. WALKER: It seems to me that those words that we used are possibly the right words that would carry that implication or requirement with them --

MR. OKRENT: And you think --

MR. WALKER: -- instead of limiting the time to have a facility, the outliers in a reliability sense, and

those things that contribute significantly to improved plant safety can be incorporated into the design.

MR. OKRENT: Now do you have criteria that you would propose for use in judgment on where it pays to make improvements? How would you go at that?

MR. WALKER: We don't have criteria today, no. But I think that once the safety goal is in place, then those [illegible] could be derived relatively rapidly from the overall safety goal in the context of that goal such that they did make sense.

MR. OKRENT: Now the Canadians, for example, have expressed a kind of quantitative reliability requirements on systems, as I understand what they have, where they try to use, I guess, one redundant system that they think they need a factor of 10^{-3} reliability, and 2, if they need something less than this, so in effect there is a kind of implicit safety goal, and then also some rule saying you can't put all your eggs in one redundant basket.

Would something like this work for your kind of LWR? I am trying to see if there is some kind of an understandable yardstick that can be applied now which the Staff might use to say: If you do reliability studies and find that you at least meet -- I'll invent a number like the Canadian number -- 10^{-3} where you would need one system, but you don't look for 10^{-6} out of one single system in that case;

you would have two, or something else, where you would cut down the frequency, or whatever.

Would something like that be needed, or useful? Or would it just get in the way of trying to pursue the process that you've indicated you used and you could do on the plant?

MR. WALKER: Well, personally I believe that a quantifiable goal is essential. And once you've talked about an addition, it seems to me it is a goal that takes into account what a system must perform in a functional system, the importance of that function and whether or not there is additional redundancy associated with that function. And to me, those kinds of goals permit one that kind of flexibility in the design which are useful and I would much like to see goals phrased in that fashion.

MR. OKRENT: Do you think the industry could propose goals like this to be used in reliability evaluations and in design that would be sort of generally acceptable to the industry and to the NRC and third parties who come in with an open mind?

MR. WALKER: I would hate to commit for the industry, but as you know there is an AIF group working on an overall safety goal, and I don't believe they have addressed system reliability goals, as such.

MR. OKRENT: I am trying at the moment, as you can tell, to stay in the area of system reliability goals, which

is what I think you need for these reliability evaluations.

MR. WALKER: I agree.

MR. OKRENT: When you get into this overall safety goal, that is related to containment and other things which are not unimportant, but if you could divorce the two, at least to the extent that's practical in doing the reliability evaluations -- I mean, you would keep in mind what the nature of the containment was and so forth, but nevertheless you wouldn't want to hinge your reliability evaluation every time on what might be the features of your containment in the next go-around.

MR. WALKER: Right.

MR. OKRENT: Well, I'm not sure you -- You weren't saying AIF is trying to compose --

MR. WALKER: Really what I was trying to say is that AIF has together a group that is considering overall safety goals, and this is possibly a vehicle by which the industry could take on the more detailed question of system reliability goals.

I don't believe that goal has considered such goals as part of their charter, as yet, but that is one possible vehicle by which industry could take on such a task.

MR. EBERSOLE: As a case in point, between the three PWR vendors, I find some interesting differences between their activity control system in the context of the scram

system, shutting the plant down. I believe B&W has introduced some interesting diversity in some of their newer designs. Westinghouse sticks to simple redundancy using DB-50 breakers for the magnetic circuit openings; where it might be really quite inexpensive to employ diversity, but we don't have a goal that demands it. But we've gotten a long way, or we've gotten some way without those goals just on a deterministic basis.

What keeps -- just in that particular area -- Westinghouse from employing diversity to enhance this function without having to have some numerical goal to force it? It doesn't cost a lot of money.

MR. WALKER: I'm just not able to answer that question, because I don't know the Westinghouse design practice or rationale associated with design of those systems.

MR. EBERSOLE: Well, you use a couple of DB-50s in parallel for breaking the magnetic -- They're absolutely redundant; simple redundancy; no diversity -- a lot of diverse inputs, but they funnel down to simpler things.

MR. WALKER: Yes.

MR. EBERSOLE: Really, there must be a counterpart of the aux feedwater study system looking at the reactivity control systems in PWRs of which of these look to be on the worst end of the scale, or the better.

MR. WALKER: Yes. I understand your comment; I just

don't know the details of the design or the rationale associated with [illegible]t Westinghouse systems.

MR. OKRENT: Okay, why don't you go on, and we may come back to this point.

MR. WALKER: Okay. I was mentioning the Standard Review Plan review to identify all deviations from the acceptance criteria. We don't believe that such a proposed SRP audit is necessary, or that it will contribute measurably to plant safety. The SRP, at least to us, was a guide or a statement by the NRC Staff to itself and to the public of how it conducts technical reviews of applications.

When it was issued, the SRP merely described the existing system, rather than creating a new or revised system of technical review by the NRC.

For the most part, the SRPs were published in 1975 following many months of preparation. Since our application was reviewed in that period and the Safety Evaluation Report was issued in September of 1975, and since the Staff review process was not changed by the SRP, we feel there is substantial reason to believe that the FNP technical review was performed in the same manner as stated in the SRPs.

Before any of the NTCP/ML plants begin operations, the entire technical review process will be repeated for the final plant designs. At that time, any question of conformance with the SRP can certainly be decisively answered.

Additionally, the SRP audit program would require commitment of substantial Staff and Applicant resources. We believe that the furtherance of reactor safety would be better served by utilizing these resources in a more productive effort.

With respect to design changes for mitigating the effects of degraded core conditions, possible hydrogen control features were discussed earlier, as well as the core ladle. If as a result of the degraded core rulemaking further design features are required to control the quantity of fission products that might be released, a containment vent has been evaluated in a preliminary way for the FNP.

This device could be multiple vent openings leading to the basin in which the FNP floats. It would contain a device like a rupture disk and would vent beneath the surface of the basin in case the containment pressure exceeded the rupture disk pressure. The basin water would serve to scrub and retain a large fraction of the vented fission products -- except for noble gases.

This concept was discussed both in our Liquid Pathways Generic Study and in our September 1979 responses to the ACRS questions. And sufficient evaluations have been performed to determine that this design approach could be installed in the FNP if it were required by rulemaking.

With respect to safety goals, we endorse and support

the use of quantitative safety goals in determining the need for design changes to reduce the vulnerability of plants to accidents. We believe such goals are essential to provide a quantitative and uniform basis for reaching decisions on the need for devices such as additional containment features.

It is just as important that the NRC Staff and the ACRS accept both these goals and a rational consistent approach to their application for design changes if such goals are to be useful.

This completes my statement. I do appreciate the opportunity to comment on the proposed NRC rule for proceeding with licensing of near-term construction permit and ML application, and hope our comments have been useful to you.

MR. OKRENT: You mentioned earlier that you do have a capability to increase the containment's pressure rate, or whatever words you used --

MR. WALKER: Yes.

MR. OKRENT: Could you tell us a little bit more about that, and how you react to that part of the Staff's list? That was an item, I think, on Mr. Myer's list.

MR. WALKER: Yes. Well, in the submittal of last year, we presented to you a diagram of our containment and indicated upon it the pressure retaining capabilities of the various regions of the containment. There are two areas that

stand out as capable of being beefed up to give additional pressure capability. One was the large equipment hatch; and the second was the upper ring on the containment, which would require some additional material be added to that ring.

You might recall that our calculations indicated that the pressure capability of that containment was about 45 pounds, and we could probably go to about 65 without too much trouble.

With respect to our containment, the problem we have -- if I can take that coffee cup and show you -- was we have a containment sitting on a steel platform (indicating), and the juncture between the platform and the containment is a design problem and an analytical problem. Our current calculated pressure capability there is about in the neighborhood of 65 pounds, and I think that is what would limit easy fixes to our containment.

MR. OKRENT: That's not at normal stresses, but at failure point, the 65?

MR. WALKER: Yes, it is. I've forgotten whether we used element, or 120 percent of element.

MR. OKRENT: Now in its action on Sequoyah, if I recall correctly, the Commission said something like, after a year there should be a hydrogen control system that allows significant margins, or something like this, with regard to the strength of the containment. And that is what they suggested

for a plant already constructed. I don't remember if my words are exactly correct, but I think that was the sense of the decision; and the other opinions were for more stringent requirements like to have normal code margins, given the hydrogen combustion.

Where and how does FNP then fall in this somewhat grey range of possible requirements?

MR. WALKER: Well, so far we have looked at the potential. We're putting ignitors in our containment. We think that is feasible and doable.

We have looked at the potential for strengthening our containments. We think if that becomes a requirement that is also feasible and doable.

We have looked at inerting systems for our containment. We think that is a horrible way to go because of need to enter the ice condenser containments frequently.

And that's really about the extent of the evaluations we've done. We are keeping abreast of the stuff that is being done for Sequoyah and for McGuire and actively participating in that effort, so that whatever those requirements eventually develop to be, we feel like we will have to add design features to meet them.

MR. EBERSOLE: Have you looked at the reasons why you have to enter the containment so frequently, with a view toward reducing that need somewhat?

MR. WALKER: Yes. We know that the problem in those containments is maintenance of the refrigeration equipment.

MR. EBERSOLE: Is any of that removeable to the outside?

MR. WALKER: Difficult to do.

MR. EBERSOLE: You've got a lot of instrumentation in there, too, haven't you?

(Pause.)

MR. EBERSOLE: Reactor instrumentation.

MR. WALKER: Inside containment?

MR. EBERSOLE: Yes.

MR. WALKER: Again, I can't really answer that question effectively because I don't know.

MR. EBERSOLE: Are you looking already at the vulnerability of those items to flash burns from the ignitor functions? That's one of the things that TVA has to face. The fast burn.

MR. WALKER: Yes. Well, my recollection is that the bulk of the refrigeration equipment is back behind the ice condenser walls, so in a sense it is shielded from the --

MR. EBERSOLE: Yes. I'm speaking about mitigating equipment, though, that needs to be kept -- you know, like air return fans.

MR. WALKER: Yes. Right. Not specifically our

plant, no, we haven't started that effort yet.

MR. EBERSOLE: That's inevitable, I guess?

MR. WALKER: Yes.

MR. OKRENT: Let me ask you a different kind of a question. Sometimes when we talk to the Staff, and in fact I think Mr. Myer alluded to this this morning, about mitigative features, there's a question raised: Well, this particular feature may help in this event, but we have to look at how it might also be harmful.

Now you mentioned a momen' ago hydrogen ignitors. I suspect one can postulate scenarios where they're helpful, and scenarios where they're harmful --

MR. WALKER: Right.

MR. OKRENT: -- depending on what you care to postulate.

How will you factor this kind of thinking into hydrogen ignitors, for example, if you're forced to look at this?

MR. WALKER: Well, if I were to take that task on today -- Let me just back up one step. As perhaps you're aware, we have participated some in the Indian Point/Zion study. As part of that study, we have developed a fairly detailed containment event tree. We have recognized the need to develop those event trees also for the ice condenser containment. We think that would be quite a powerful tool in

permitting us to identify events where particular mitigating systems help you and where they harm you, and eventually quantifying the benefit, both plus and minus, you might get from such devices.

MR. OKRENT: Is that kind of help and harm done for other kinds of safety features? We have certain features on the plants that are intended to prevent serious accidents from occurring. They might also be the cause of a serious accident. I suppose maybe the accumulators which are there to handle a large LOCA represent some kind of connection that can rupture, and it's a high-pressure vessel that can rupture, and so forth. I mean, there are positive and negative features to almost everything on the plant. Even the control systems have positive and negative features, and so forth.

Do you go through the plant and look at each feature and try to see what its positive and negative aspects are the way they're now starting to do on things like vented filter containment, for example?

MR. WALKER: As far as I know, that exercise has not been done in any of the risk evaluations that I'm aware of.

MR. EBERSOLE: Overhead injection has been done, in part.

MR. WALKER: Yes, that's true. We looked at upperhead injection with that in mind.

MR. OKRENT: But I suspect they're only in an

incomplete way. I don't think there was a real balance sheet made with all the possible failure modes that might go with the accumulators, and so forth, for the upperhead injection system. There is a partial list of scenarios.

MR. WALKER: Well, at least in our case, it was more of an attempt to identify what some of the negatives might be in a qualitative rather than a quantitative way.

MR. OKRENT: In other words, I guess one can look at what people are doing on some of these mitigative features in two ways.

First, that maybe they are blazing a trail that we should be doing for all systems. Or maybe they're asking perfection there where we don't ask perfection anywhere else. Namely, that it be only good and not bad; that the policeman's gun can only help prevent crime, and never kill an innocent person.

MR. WALKER: Yes.

MR. EBERSOLE: Let me ask a question on core cooling. On core cooling, you eventually come to a cross-roads. Are you going to guarantee, without fail, the presence of a secondary system for heat removal? Or are you going to ultimately decide you may invoke bleed feeding?

MR. WALKER: Gee, we really haven't -- I'll pass on that one.

MR. EBERSOLE: Okay.

MR. WALKER: As you recognized, we have a Westinghouse NSSS and we will undoubtedly go with the Westinghouse position on --

MR. EBERSOLE: Well, had you ventured and said: Well, we're going to count on bleed/feed as a mitigating function to the secondary failure, I would have then asked you: What are you going to do with the PORVs? And why don't you find some good valves, or ways to keep the primary loop open --

MR. WALKER: Yes.

MR. EBERSOLE: -- rather than these compromise ways which we now have, so you can pump water through it. You have a very poor way of trying to pump water through the primary loops in a PWR. It's impeded at all points. You've got to have a LOCA, really, to do it.

(Laughter.)

MR. OKRENT: Any comments from the Staff on what Mr. Walker said?

MR. PURPLE: None from me, except perhaps to -- we may have sounded fuzzy on reliability analysis; but what I heard Mr. Walker describe they were doing sounds very much like what we had in mind.

MR. OKRENT: Maybe it was me. Let me go back and read what this was you have in your --

(Pause.)

That you say is: "The Applicant should perform

simplified system reliability analyses of a certain system." Then you say, "The Applicant should provide sufficient information describing the nature of the studies, how they are to be conducted, the completion dates, and the programs to assure that the results of such studies are factored into the final designs."

Well, I think those words are just enough in the middle that they don't say that the designs have to do anything. On the other hand, they say: But we haven't ignored the question. This term "factored into the design" is -- I don't know how to interpret it, frankly.

MR. PURPLE: I see.

MR. OKRENT: Maybe it's all clear to you.

MR. PURPLE: So that it may be, and obviously is, a fault of the words. I was merely pointing out that what I had heard described is what we had in mind. You're looking for outliers, and you're looking for them in time to influence your final design, not after.

Certainly we can take a relook at those words and try to make clear what they are.

MR. OKRENT: I mean, have you thought about when you have to look at see whether you have a potentially undesirable interaction among systems? If you want to correct it, it seems -- there's a certain point, it seems to me, when many changes are impractical because you just have too much

concrete and steel already in place, and no space-- even if you wanted to add another system -- and an inability to change dependence on a single header, or whatever it is.

So that's why I said that I find this wording sufficiently in the middle to be interpretable by everybody either the way he likes it, or the way he doesn't like it.

MR. PURPLE: We'll try to make it clearer.

MR. OKRENT: And it doesn't give very much guidance to the utility as to at least how the NRC will judge what constitutes sufficiency or adequacy. I'm not trying to excuse the utility. I assume you can tell. But I don't think this provides very much guidance.

If you have a way of making it clearer -- if that's the right word -- I would encourage you to do so before Friday? When do we meet?

MR. PURPLE: Friday.

MR. OKRENT: It would be helpful to see if you have any further words in this area, because it could be useful.

MR. PURPLE: Okay.

MR. WALKER: Okay?

MR. OKRENT: Thank you.

By the way, if I could ask a question that arises from this discussion -- and I will welcome comments from representatives of other utilities, as well as Mr. Walker, and

the Staff: I have heard it suggested -- and I don't really know whether it's so or not -- that if a utility were looking at the plant it had proposed in the construction permit and decided that in fact it could be meaningful to make a change in design -- let me invent a change which I am pulling out of the air -- they had three steam-driven auxiliary feedwater systems, and pumps, and now they're going to go to two steam and two electric or something, and that may be a poor example, but something at least that not only is different from what they had, it may be different and in their mind an improvement over what anybody has, that if they bring something of this sort in, the Staff has to go through a fairly extensive review and it sort of opens up the -- maybe not only this part of the plant, but the rest of the plant for reevaluation, can impose fairly long licensing delays at one step or another, and that this acts then as an inhibition on proposing changes.

Now I have heard this suggestion; I don't know if it is a real situation. And my example which I have just invented now is probably a very bad one, if it is a real situation.

Can anybody comment on this?

MR. SCHWENCER: Well, of course the ignitors on the ice condensers sort of fits that, Dave. It was a proposal to deal with the issue, but one that we couldn't readily grapple with. So I think the answer is, at least from my vantage point,

everything you said is true. You're likely to run into, if it is proposed, it has to be looked at to see the negatives and the positives. You don't just want to dump it in without having some proposal -- assurance that the proposal has had some independent technical look-see by the Staff.

Hopefully, the argument is made so well that the person can just go through and check it off and say: Yes, yes, I agree. But it does have to be looked at. I think I would have to hold that position.

MR. EBERSOLE: In this context, isn't it true that the Staff is probably the main obstacle to the Applicants coming in with a dedicated shutdown heat removal system, which is probably the most significant improvement you could make?

MR. PURPLE: Why would we be the reason for them not coming in?

MR. EBERSOLE: Yes.

MR. PURPLE: I say, why would that be?

MR. EBERSOLE: Because it would open up new areas of investigation.

MR. HAASS: Rocking the boat.

MR. EBERSOLE: Rocking the boat, inviting possible cancellation of requirements elsewhere which the Staff would be unwilling to cancel.

In other words, the prevailing structure inhibits

improvement, for the very reason that Dave brings out.

MR. PURPLE: You might also say that the structure for that same reason prevents a downturn when you thought it was up.

MR. EBERSOLE: Yes, but it perpetuates mediocrity, and some way has to be found to break that.

MR. HAASS: It's one of the products of standardization.

MR. EBERSOLE: Well, it could be, unless we had a good standard. Standardization can be perpetuation of poor practice. On the other hand, it can be innovation and establishment of good practice. So you can't say it generically. It has been used in the former, not the latter sense.

MR. OKRENT: Yes?

MR. WALKER: A comment also with respect to the hearing process. I believe that tends to inhibit one's desire to make any proposed changes for fear of reopening the hearing record on a particular issue that they would like you to have closed.

(Laughter.)

MR. OKRENT: Okay. Well, if anybody can add further light to that point, we would be happy to hear it. I think Mr. Robert Myers is here to make a presentation.

MR. MYERS: I am Robert V. Myers. I am the

Vice President of Generation Resources, representing the Puget Sound Power & Light Company.

I am grateful for the opportunity to appear before this subcommittee to present Puget's views on the status of the NTCP licensing situation. I would like to caution you that I am here to discuss general policy, and don't have with me a technical staff, and so we're not prepared to go into the details of what we do or do not intend to do on specific systems, or reliability criteria, or anything else.

I do understand that in the letter that you received from Houston, they indicated that they would be prepared in February to talk further to some of the issues that I think you might like to hear from the utilities on some work they have done in that area.

As you are aware, the six NTCP member utilities have six construction permit applications for a total of eleven nuclear units still pending before the Nuclear Regulatory Commission. The projects in question represent an imbedded investment in excess of $1 billion, and a total potential investment upon completion of over $45 bill .

Further, they will provide more than 13,000 megawatt electrical of vitally needed capacity initially expected to be available during this decade. In fact, our plant was initially expected to go into operation in 1981.

Although the projects may differ in design, they are

all affected by the uncertain licensing climate. Little of any significance has occurred on any of the dockets since March 28th, 1979, the date of the accident at Three Mile Island.

As a consequence, each utility is now being forced to decide whether or not to continue with its nuclear project or to look to an alternative form of electric generation with amore stable regulatory path. The life of the NTCP projects requires the receipt of clear, definitive, regulatory criteria today.

On September 25th, 1980, the NRC issued its proposed rulemaking for "Proposed Licensing Requirements for Pending Construction Permit and Manufacturing License Applications." The proposed rulemaking suggested the following three possible options for licensing action relative to the pending NTCP plants:

Option one was to resume licensing using the pre-TMI CP requirements augmented by the applicable requirements identified in NUREG-0660, and as delineated in NUREG-0718.

Option two was to take no further action on the pending applications until the rulemaking actions described in the Action Plan had been completed.

And option three was to resume licensing using the pre-TMI CP requirements augmented by the applicable requirements identified in NUREG-0660, and to require certain additional

measures or commitments in selected areas -- for example, those that will be the subject of rulemaking.

The near-term construction permit applicants submitted extensive comments on the rulemaking, including addressing the particular requirements of siting, degraded core rulemaking, reliability engineering, emergency preparedness, the remaining issues embodied in NUREG-0718, and deviations from the Standard Review Plan.

In those comments, we urge an adoption of Option 1 as stated in the rulemaking -- Namely: "Resume licensing using the pre-TMI CP requirements augmented by the applicable requirements identified in NUREG-0660."

Option 1 has in effect been applied to existing construction permit holders because facility construction under those licenses has not been suspended and the requirements of NUREG-0660 are being implemented during the course of post-construction permit and operating license licensing review for these projects.

Option 1 for the near-term construction permits provides a proper basis at this point for construction permit licensing because it carries with it a commitment to comply with NUREG-0660. With the implementation of the many NUREG-0660 requirements, substantial and sufficient improvements will be achieved for the near-term construction permit plants.

The NRC Staff's, prior to today, preferred course

for the proposed rulemaking was Option 3 -- "Resume licensing using the pre-TMI CP requirements augmented by the applicable requirements identified in NUREG-0660 and require certain additional measures or commitments in selected areas -- those that will be the subject of rulemaking."

The difficulty with Option 3 is that it does not clearly and adequately reflect a definitive licensing path which bounds requirements appropriate to a construction permit proceeding. In addition, there is no technical basis for distinguishing between projects with construction permits and those near-term construction permit utilities without.

I am going to deviate from my remarks here for a minute to stress that point. I think many of us are in this situation with a sort of a "luck of the draw" or the inadequacy of any one of a number of things, including perhaps support during the hearing process, the lack of control under the intervention process, administrative -- or the adversary hearing process, any one of a number of things. To decide that you've got us set aside here and held captive, and now dump the problems of the industry in general upon us and have them cured on these six construction permit applications, it seems to me is not only unrealistic and unfair, but unwarranted.

I also must take exception to the inclusion in the rulemaking notice of the statement that "it would be relatively easy to provide design flexibility to implement

potential significant safety improvements." Substantial design changes and backfitting during construction have an adverse impact on overall project cost and schedule, and seriously undermine utility financial resources and diminish public confidence.

Thus, the proposed rule as embodied by Option 3 requirements does not provide the definable and realistically bounded guidelines necessary to create a stable licensing process.

It has been 21 months since the TMI-2 incident. Action taken by licensees and the industry in common, as well as the technical findings which were the outcome of various TMI-2 studies and investigations, have in essence already accomplished objectives of the proposed rulemaking -- namely, significantly and cost-effectively reducing risk.

The signals from the NRC are confusing and contradictory. In the face of such signals, the NTCP companies cannot make prudent decisions either in managing the capital with which they have been entrusted or in providing service for which they are franchised. This poses a nuclear dilemma that is fast approaching nuclear paralysis. Utility executives have no sound basis on which to decide whether to continue with further commitments to our nuclear projects.

I would like now to make a few brief remarks about the need for power in the Northwest and our service area in

particular, as well as the importance of and our reliance upon the nuclear option for vitally needed electrical generation.

(Slide.)

One of the reasons for me doing this is that there was a mischaracterization of the situation that many of us are in before the Full Committee of the ACRS which said that it really didn't make any difference whether you licensed any plants in the next 10 years or not because they didn't really need the power anyway.

Puget Sound Power & Light Company is the largest investor-owned utility in the State of Washington. The company serves approximately 1.2 million people within a 4500-square mile service area which includes 8 of the fastest growing counties bordering Puget Sound in Western Washington, and Kittitas County in Central Washington.

Puget Power's generating capability is currently about 79 percent hydroelectric, although all future major supplies of electrical energy will be produced by thermal generating projects. I might note that on this particular slide, we still show the Skagit site located in Skagit County. That option has also been foreclosed as a result of time passing, as it inevitably does, and the rezone contract which we had which authorized the use of that site for a nuclear power plant expired on December 1979, which at the time we entered into that agreement seemed like a fairly safe distance

in the future, which was five years earlier.

(Slide.)

The energy situation faced by the Western States, and in particular the Pacific Northwest, as we head into the 1980s is critical. The extent of the energy problem in the Northwest is indicated on this next figure, Figure 2, which is based on a summary of estimated loads and resources as of May 1st, 1980 developed by the West Group Forecast, a compilation from utilities in the Northwest Power Pool.

The most important conclusion from this figure is that there are firm energy deficits in all years through 1991 under adverse water conditions. The dark line (indicating) is the load, and the bar are the resources. Now it stops in 1991 because that's the span of time covered by the Northwest planning system. It doesn't mean that in 1991 the problem solves itself. In fact, in 1991, the resources which we are now counting on to meet these loads are again the nuclear plants which 10 years earlier were to meet the loads in 1981, the Skagit project and the Pebble Springs project.

Since 1975, the number of customers we serve has increased by nearly 30 percent, including nearly 30,000 new customers added in each of the years 1978 and 1979.

(Slide.)

The outlook for the '80s is for continued growth. The Washington State Office of Financial Management has predicted

a population increase of 33 percent for the Puget Sound Region during the next 10 years. Responding to such growth presents a challenge of considerable magnitude, particularly in light of the energy uncertainties of the '80s.

The plain fact is that our ability to meet the growing demand for power in our service area is being stretched thinner with each passing year, primarily due to excessive delays in bringing new generation resources on the line.

(Slide.)

The very substantial delays in nearly all major power plants planned in the Northwest for the '70s and the '80s are graphically illustrated in Figure 4. Before I go to Figure 4, I just should show you the resources and the load picture as it is covered, looking at the resource mix that we have.

(Slide.)

You can see that the principal part of our supply is provided by hydro. We have a substantial amount of coal. We are planning now for nuclear that will come on line. The nuclear (indicating), which is this section of the bar over here, which stops at this point (indicating), is the N Reactor at Hanford, whose operation is currently scheduled to terminate at this point (indicating). Hopefully it can be extended, but currently it stops then. And we pick up again with a small portion of nuclear energy, which is Puget's share of the

Washington Public Power Supply System Plant No. 3 located at Satsop, Washington.

MR. EBERSOLE: Is that estimated load based on rising cost of energy, or present cost of energy, or what sort of base?

MR. MYERS: It is based on our best estimate of what the future will hold. It includes adjustments for the price elasticity. It includes estimates for contributions of conservation. And it also reflects the changes that we have seen in the near past where substantial changes in consumption patterns have occurred.

We are projecting on this line (indicating) about 4 percent per year growth. Historically we have seen something on the order of 8 percent, a substantial change from what we have seen in the past.

The very substantial delays in all major power plants--

(Slide.)

-- somebody once observed there are eight ways to put one of these on there, and sometimes you can go through seven before you get it.

(Laughter.)

MR. OKRENT: Or you can go through nine, because you repeat some of the previous error.

(Laughter.)

MR. MYERS: I did that in three. I thought that

was pretty good.

(Laughter.)

MR. MYERS: The very substantial delays in nearly all major power plants planned in the Northwest for the '70s and '80s period are graphically illustrated on Figure 4. All the plants shown are nuclear, with the exception of Colstrip Units 1, 2, 3, and 4, which are coal-fired plants located in Colstrip, Montana.

I should add that we now are considering the addition of 2000 megawatts of additional coal-fired capacity called the "Creston Units" located 60 miles west of Spokane because of the problems which are obvious in this graph.

Clearly the Northwest is heavily dependent on the nuclear option for future generation. If further delays beset these planned nuclear porjects or if continued licensing uncertainty forces lengthy deferral or cancellation of these projects, the economic, environmental, social, and institutional impacts on the Pacific Northwest and Puget Power's service area will be disastrous.

In case it isn't clear from the chart, the left end of each of the bars is the originally scheduled operational date for the project. The right-hand end of the bar either represents the realization of operation -- which in the case of Trojan, Colstrip 1 and 2 has occurred -- or, our best estimate of when commercial operation of these units will occur under

current conditions. "Current conditions" do not include some of the assumptions that one might make after spending the morning in this room, I might add. They are more optimistic than they will be tomorrow.

In summary, I strongly urge that a clearly defined licensing direction be established for the projects of the near-term construction permit group. That direction must not take the form of continued requests to commit to the results of open-ended proceedings that are not likely to be resolved for some time.

To make the construction permits contingent upon the resolution of these issues dilutes the value of the construction permits while leading to ever increasing and unbounded costs for the completion of the projects.

Rather, I propose that we be allowed to go forward on the realistic course specified by Option 1 of the rulemaking. Only through a commitment by the NRC to clearly define the licensing path and safety requirements can the near-term construction permit companies expect to gain needed support from our directors, shareholders, and ratepayers to continue with the nuclear option.

This concludes my comments. There are representatives of other NTCP companies in attendance, and they could respond to questions on issues that might be particular to their own situation.

MR. OKRENT: If I could explore a few general questions, how technical is it fair for me to get? You tell me.

(Laughter.)

MR. MYERS: All right. You ask the question, and I'll tell you whether it is beyond my expertise.

(Laughter.)

MR. MYERS: That seems like a safe place for me to be.

MR. OKRENT: Fine. Okay, now when we met in May when the ACRS quite consciously held a subcommittee meeting to find out what was happening on the NTCP plants, and representatives of the utilities were in, the utilities then indicated that they would try to come in with proposals for how to develop some kind of policy approach to the items then in need of resolution. There were six on that list, you may recall. It appears that the number of issues of concern have decreased, for one reason or another.

In what I read of the response of the utilities to what the Staff published for comment back whenever they did it, in September, or the summer sometime, I didn't see -- I may have missed it -- much in the way of a proposal from the utilities for what to do, other than I guess what you call Alternative 1, which was really to proceed on the current basis and let's see, I suppose, what the rulemaking procedure

will hold in the future with regard to backfitting.

I had thought, from the meeting we had in May, that there might be some kind of alternate other than that that the utilities would propose that they thought, I suppose you might say, met the spirit of whatever was relevant in the safety area, and would provide some kind of a suitable basis which might in fact give both the utility and the NRC a basis for saying, "this looks like a reasonable proposal; in fact, maybe we can, barring anything unexpected, even grandfather these from the rulemakings," you know. That would be more stable than your sitting with a construction permit, but not knowing when the shoe is going to drop -- one, or two, or three -- from rulemakings.

Now --

MR. MYERS: I was not involved in those earlier discussions, except that they did occur early enough, and subsequent to that there were meetings that were held among the NTCP owners and the Staff, and I don't think much progress was really made.

The proposed rule came out. It contains what it contains. We view that as asking us to go ahead and build something that has no finite limits to it physically, so that you don't foreclose your ability to put anything inside or outside or across any boundary, barrier, system that might be later decided you should have.

In addition, it seems to say that you are uniquely different than the ones who we just issued construction permits to, even though you might have a design that's further along. Our design is 70 percent complete, in terms of engineering. We certainly have had plenty of time to do it, and I'm surprised it isn't 100 percent complete, except for changes in the requirements.

So we find ourselves in a position that it makes it very difficult for us to propose something when we have no clear set of objectives that anyone has said: This is what we're trying to do. We are in the position of the -- a friend of mine, which talks about the fellow who is asked to "bring me a rock." And with the large number of rocks available for him to select from, and you being the only individual who knows which rock you want, he can make a lot of trips before he ends up with the right rock delivered.

So that is really the situation we are in. We don't know what we're trying to do, in terms of safety criteria. What standard are we trying to meet? What are we trying to achieve through this? What difference is there between us and the other plants that are going ahead now with a different set of criteria? I think we're just bewildered at this point. We have no basis to optimistically view the future and move forward, spend more money. We have $270 million, and some of it was very good dollars back in 1973, invested in this plant.

It's really gone.

MR. EBERSOLE: Well, TMI-2 happened. TMI-2 happened. Suppose there was no regulatory pressure on you to do anything. TMI-2 happened. What would you have done? Would you have gone back and examined your aux feedwater reliability? Have you, yet? Have you looked at anything else like that in these eight dozen critical systems?

MR. MYERS: We are a BWR-6 Mark III.

MR. EBERSOLE: Well, okay, the counterpart of those problems.

MR. MYERS: The industry is looking at the problems resulting from TMI. We've got the Institute for Nuclear Power Operations. We've got the other organizations trying to look at the sorts of things that led up to that -- the operator error sorts of things, the limits on systems, the transients that can get you into trouble. We've now learned that you're better off looking at transients and perhaps smaller events than just concentrating on LOCAs, which fascinated us for years and turned out to be the wrong thing to be fascinated with.

I don't think the industry is trying to shirk its responsibility, but we do have to know where it is we are headed. We are in a race where the finish line has not been told to us yet. We're just told to keep running. We do have limits to our ability to do that. We have financial limitations. These plants cannot be constructed regardless of the cost. We

need to know -- frankly, our utility needs to know if we should throw the $270 million we have invested in this plant, just throw it into the system and forget it, and pursue some other option. Because we cannot define what our ultimate exposure is with the system we have.

MR. EBERSOLE: Well, to some extent you were always in that position; only now, you think you're worse. I mean, you never knew what was ahead of you with the evolving requirements.

MR. MYERS: To a degree, I suppose that's so; although there is some comfort in having at least more than two or three other people in the pool with you. We are being really set aside here as such a small group, without a lot of people who are going to rush to our rescue, because not a lot of people get impacted with us. That is a reality of the world.

MR. OKRENT: Well, there are two aspects of the discussion that are maybe worth looking at a little bit.

I guess with regard to your example of "bring me a rock," I had hoped that perhaps the utilities were going to say: This is the kind of rock that if we get it, it should provide an acceptable basis for these plants, and here is why. And I guess that's what I thought we might see forthcoming after the May meeting.

It is not clear to me at all that that's not a fair action to expect of a utility, or a group of utilities --

in other words, that they leave it for the NRC. And I understand in fact there may be some utilities, or maybe the group are trying to do that, but at least we haven't seen that yet.

So in any event, I think, while you've heard us give the Staff a hard time here, and before this, in not providing a somewhat better delineated guidance, I think that could equally well have come from the utilities saying: Look, we think there are improvements that are relevant, or here are bases for making improvements that we think we both agree we can proceed, and so forth, and should provide an acceptable basis for these plants.

I don't think it's fair to assume that these plants, you know, aren't even on the drawing board, and no effort ever existed. On the other hand, I think you do have to accept reality. An accident did occur. People have taken a look. In fact, they are using information that was there before, but now they are giving more credence to it, more weight, and maybe they are judging that in fact there are things that are more probable, or that need remedies, or warrant remedies, whatever the situation is.

So I think, myself, the NTCP plants are somewhere in the middle. They're not like the plants already in operation or under construction, and they're not like plants that might begin design in a few years. There is precedent, I

think if you look elsewhere in society. If a dam fails, a dam just ready to begin construction is relooked at, and in fact sometimes even stopped. It depends on the circumstances of the dam. Some go ahead. And similarly in other situations.

I don't think INPO, to my knowledge, is addressing the design-related questions. If they are -- and I didn't know that NSAC was. Maybe they will in the future. But there has been a period of time, which is getting on two years, when in fact some of these things might, I think, have been initiated by the utilities themselves whether the Staff was doing something or not. Maybe I am a dreamer, but correct me if I am wrong.

MR. MYERS: I don't know how to respond to that. I think the utilities are trying to be responsible and to be responsive, but you've got to remember that we grew up in a system that was created for us. We didn't create it. We tend to be reactive.

We also tend to be smaller organizations with limited resources. I probably don't even understand today all of the things that are going on in the industry, and I am sure that General Electric, the vendor of our reactor, has done many studies that relate to the aftermath of TMI and systems that are contained therein.

I didn't come here today prepared to talk about those, but indeed if your charge is that we didn't come in with

a proposal to change the licensing process of the Puget Sound Power & Light Company, then we're guilty; we did not.

MR. EBERSOLE: How many BWRs are you talking about?

MR. MYERS: For Puget?

MR. EBERSOLE: Yes.

MR. MYERS: Two.

MR. EBERSOLE: Just two?

MR. MYERS: Our first two reactors, if they are built.

MR. OKRENT: Mr. Lowenstein, did you want to make a comment?

MR. LOWENSTEIN: Yes, just a small one, just to supplement what Mr. Myers has said.

I'm not sure he's familiar with the transcript of the meeting to which you have referred, Dr. Okrent. I think part of the answer to your question is that time has passed that discussion by. Much of that discussion, as I remember it -- and I haven't reread it since the time of the meeting -- was in terms of what might be proposed by utilities to deal with subjects such as prevention, or mitigation of degraded cores.

Since that time, the Commission has issued an advance notice of rulemaking; it has issued a proposed interim rule; there is a great deal of activity going on both within the Commission and among industry groups in the context of the rulemaking. And I think the discussion that we heard this

morning makes perfectly obvious that the Staff really wouldn't be in a position to act meaningfully, even if we could develop an industry position at this point on any particular proposal, outside the context of the rulemaking.

Part of our effort in trying to obtain meaningful licensing criteria for the NTCP groups has been to get criteria which would rule out a specific case of degraded core considerations, put them into the rulemaking for the time being on the basis of the interim rule or as it may be modified and finally adopted by the Commission.

So I think that some of the specific things the industry might have come forth with before will be done, but they are going to be done, I expect, on a schedule and in a framework of the degraded core rulemaking.

MR. OKRENT: Well, since we have the benefit of your comment here, could we explore with both of you: Suppose in fact the Commission adopted Alternative 1, but you knew that there was going to be a rulemaking on degraded cores and molten cores, and it might end up saying that BWR-6 Mark IIIs have to somehow cope with large amounts of hydrogen generation, and molten cores, and so forth and so on; but the decision came three years from now, or five years from now, or X years from now, whenever it was that the Commission arrived at a rule.

Now would that leave you in any happier position

than the Staff's Alternative 2, 3, or 4? You remember today that they had 2, 3, or 4.

MR. MYERS: Oh, yes.

Well, it does, I think. There are two things that are important there.

First of all, if that were to be the case, we would find ourselves in the same position as Grand Gulf, Perry, some other BWR-6s, who would be faced with the same thing. We would be proceeding ahead, but we know that what we have here is a problem that is going to be imposed on the industry because of some technical basis for its requirement, as opposed to the fact that we happen to be handy and caught in this malaise.

Secondly, we don't presume, or don't assume automatically that we would proceed ahead; but we at least would then have the information that is necessary for us to say: Well, let's take a look and see what sort of a risk we perceive this to be, what sort of bag are we holding? And then if our evaluation is that that bag is too big for one utility of our size to hold, then perhaps we cut ourselves loose from the enterprise and go on to other things that productively produce generating capability.

But we are in a position right now, we don't even have a very good feel for what it is we are looking at. So yes, we would be happier with that.

MR. OKRENT: You mentioned Grand Gulf. It happens

that in a letter dated December 9, 1980, they wrote Mr. Tedesco of the Staff, outlining what they were doing in response to his letter dated October 30, 1980. They talk about a program for looking at hydrogen control, and it includes a range of things including many different concepts.

Now I would assume that if design measures evolve out of such a study that look beneficial, if anything they would be easier to include in a plant on which construction is about to begin, or going to begin in a year, or whatever, than on Grand Gulf, which memory tells me got a construction permit some years ago and presumably is fairly well along. I did visit the site a large number of years ago.

MR. MYERS: I would assume that is a fair statement; yes.

MR. OKRENT: So in other words, I guess I am trying to see: Do you think it is unreasonable that Puget Sound look at its proposed reactors as hard or harder than Grand Gulf? I am trying to understand what it is you are saying.

MR. MYERS: We seem to be rather argumentative in this whole process. The question that was asked earlier I think bears repeating. That is: What is the impact of all these things? If what we are doing is attempting to uniquely, on our docket, make an independent examination of our reactor and its facilities, we then are subject to the second-guessing of the intervenors and the Staff, and anybody else who cares to

participate in saying, "Look, your look really wasn't the kind of look we had in mind. Grand Gulf came up with these results. Why are they different? Why are they not comparable?"

And we are going to go through the kind of protracted kind of time-consuming adversary hearing process that has not really been all that constructive. It has to be done on a generic sort of basis. If you are going to take a look at these things, it should be done in such a way like reliability where we all agree that these are the rules, this is the way we're going to do the analysis, this is the criteria we're going to use to judge the result, and then we can move forward with it.

Then, we have something to found our belief on that we should proceed or not proceed. But to say: Each of you do your own thing; come up with your own criteria; set your own standards; make your own proposals, invites a chaos which has brought us to where we are today, which is the same place we were seven years ago.

MR. EBERSOLE: It sounds like you're endorsing a good standard plan.

MR. MYERS: Well, we bought a GSAR. We really thought we had one.

(Laughter.)

MR. OKRENT: Are there any other questions?

(No response.)

MR. OKRENT: I guess that's it. Thank you.

Are there other comments that representatives of NTCP utilities would care to, or are willing to make, as the case may be?

(No response.)

MR. OKRENT: Well, I am going to suggest that we take a break. We will come back in 10 minutes or so and talk further with the Staff, and so forth.

(Recess.)

MR. OKRENT: We will reconvene the meeting.

I guess I would be interested, if I can, in exploring a little bit the differences between Option 1 and Option 3. For purposes of the discussion, we can assume we are talking about the August version, recognizing that the Staff has some modifications in Option 3 now.

In what the Staff said in NUREG-0718 that was issued in August, they discussed Option 1 and stated that it would minimize review of construction impact, thereby minimizing delays in reaching regulatory decisions on these applications, and then they went on to state: "The principal disadvantage of Option 1 is that it fails to take advantage of the fact that since construction has not started, it would be relatively easy to provide design flexibility to implement potential significant safety improvements."

Now we have heard a range of comments about Option 3

as proposed then, and there are comments in the written material. They relate somewhat to difficulty in the hearing process, and somewhat to a lack of knowledge of what will be the NRC requirements aside from the hearing process.

First I would like to understand better in my own mind: Is there really a major difference in the long-term stability? By that, I mean not only in getting a construction permit, but really in getting an operating license, and even to years beyond, between Option 1 and Option 3? If one doesn't know what will be the outcome of, say, the IREP program, of the hearing on degraded core cooling, and so forth, why is it felt by the utilities that Option 1 provides substantially more stability than Option 3?

Can the utilities help me in this regard?

I will welcome as many participants as are willing. Mr. Myers?

MR. MYERS: First of all, I would like to say that my interpretation of Option 1 versus Option 3, Option 1 recognizes that as a result of the rulemaking there may be additional requirements imposed upon those people who have reactors of different types, across the board.

Option 3 says that I am required to do something that nobody can tell me how to do. That is, to not foreclose my ability to incorporate anything anybody might later decide should be incorporated on those few plants which were charged

with the responsibility for not foreclosing the ability to do that. And that's not many of us. That's just a couple. I don't know how to do that. I don't even know how to understand what's involved in attempting to do it.

MR. OKRENT: Let's hear the Staff comment to that specific point before we go on. Are they asking this difficult flexibility? Or may they be asking unknowingly that this be there? Or just what is it that they're asking, do they think?

MR. PURPLE: Well, I think Option 3, as it was in the draft that went out in September with respect to degraded core, one of its faults and one of the big disadvantages is just what Mr. Myers said, that it was not well bounded and would be difficult in litigation, and would be difficult -- we didn't have a clearcut understanding of what we wanted. It was written with a good deal of flexibility in the language.

I think as we're reacted to the comments and thought about it more ourselves, we have been trying to seek a method that still has what we think are the advantages of Option 3, but that are more precisely defined.

For instance, Alternative 4 that Jim Myer talked about, is a rather precise list of things to do. It's well understood. What may not be understood is what the final answers are going to be out of that, but at least it is bounded. And if the concept works, the thing could be grandfathered from the ultimate rule, and that's a well-defined

thing.

So I guess I am saying that I probably agree with the frustrations expressed in the sense of how it was drafted in September, and we are trying to fix that by seeking a way that is a better defined method.

MR. OKRENT: And you think the Option 4 that we heard presented here is or can be made a defined method?

MR. PURPLE: Yes.

MR. OKRENT: That doesn't leave the question posed open? In other words, that one has to be able to design to maintain flexibility for any eventuality?

MR. PURPLE: I think it's a whole lot better defined than simply leaving it open to whatever the rule may develop.

MR. OKRENT: Do we have written down what you consider to be Option 4? It's not in the draft. Do you have something written down?

MR. PURPLE: Not really.

MR. JIM MYER: Not exactly as it was presented, no. It is presently in draft form.

MR. PURPLE: That's why it was blank in the draft you got today, because the words have not yet been finalized.

MR. OKRENT: Do you think that by Friday Option 4 might look differently? Is that what you're suggesting?

MR. PURPLE: I guess that's certainly possible.

MR. SCHWENCER: I would expect we would have it

written down Friday.

MR. PURPLE: But it's going to be written by then, because our Commission paper is due to go to them by, say, Thursday night, so we will have to have it finalized. So by Friday, we will have it written down. It is certainly possible that it could be something different.

MR. OKRENT: Do we have it written down?

MR. SAVIO: I have what I have written down.

(Laughter.)

MR. SAVIO: I have what I heard written down.

MR. OKRENT: You do?

MR. SAVIO: Yes.

MR. SCHWENCER: Hopefully, it is within the transcript.

MR. OKRENT: But I can't remember when I said the transcript will be available.

MR. SAVIO: It will be available in one day.

MR. OKRENT: In one day?

MR. SAVIO: Yes.

MR. OKRENT: I find transcripts are not always completely intelligible -- No offense to the reporter -- it is just the way it works out.

(Laughter.)

MR. OKRENT: Yes?

MR. PURPLE: There is another point I want to make

that has come up quite often from Mr. Myers about the fact that these NTCPs are a unique group and are being treated in a fashion different than others who had luckily just gotten their CP just before TMI.

One of the things that we will be doing next Tuesday is presenting to the Commission, and seeking their approval to our plan to take all of 0718 and apply it to all other CPs that fit a certain definition. That is, CP-holders that have had CPs for some amount of time and for which very little construction has begun, to see if it doesn't make sense that some of these same requirements should be imposed on them.

I can't tell you what percent construction; we haven't figured that out yet; or how long the CP would have been held. So I don't know how big the population will grow. But we will be presenting this as a proposal to the Commission next Tuesday.

MR. OKRENT: You would propose an increase and share the misery?

MR. PURPLE: I didn't want to use that word, but that's --

(Laughter.)

MR. PURPLE: -- but we recognized the same thing. It was hard to rationalize why a plant that, for example, had already gotten a CP on March the 25th of 1979 should somehow be totally excluded from at least the same consideration of

doing the same kind of option looking that we're requiring of the pending CPs.

MR. OKRENT: Mr. Myers, did you hear enough about their option 4 to comment on whether you think it would be bounded and specific, at least not open-ended in the sense you earlier defined?

MR. MYERS: Well, I'm like you. Not having had the benefit of seeing anything written down, and commenting on a draft that I just listened to tentatively, the thing that appears to me is that the rather optimistic appraisals of the time required to do the things that were being discussed in that option probably don't lead to something within a year. Probably you're talking about a time frame that's on the same order of magnitude as the rulemaking, prior to anybody getting to the point where you say you're now ready to issue a construction permit. And so we're talking about much longer delays, again, associated with those people who don't have their construction permits.

If I recall what was said there, you're talking about going through a process that he said would take a year, which I remember when licensing took 14 months, and that was in 1973.

MR. OKRENT: I remember when it took less.

(Laughter.)

MR. MYERS: And that was a forecast for us.

(Laughter.)

MR. OKRENT: No, I said I remember when it took less.

MR. MYERS: We don't really know enough about it. I don't know enough about it to say much more than that, although I do believe that it does not provide us with a mechanism whereby we could, in an orderly way, get a construction permit to proceed ahead, and then look back and assume that we would be impacted with the same thing that others with construction permits have. I don't think anybody is so naive that he assumes, as was said here earlier for instance, that anybody could grandfather you against further decisions by anybody.

We all know we're exposed to that as a justification for a modification or an additional system is made, and we're going to be hit with it if you have one of these plants. We know that. But at least you get to start and get going and proceed ahead, and you know that when it's looked at and its cost effectiveness is measured across the industry. Is this a change that's worth making? And we don't find ourselves two years from now with this much longer delay, that much more money invested in just the cost of the money we've got invested in the plant, and then the same excuse being used to say: Well, they haven't started construction yet, so why don't you go ahead and lay this one them?

MR. OKRENT: Well, now, I have been pushing on the Staff; now let me push on the utilities. The Staff in their document said that the principal disadvantage of Option 1 is that it fails to take advantage of the fact that since construction has not started, it would be relatively easy to provide design flexibility to implement potential significant safety improvements.

Certainly that's true for certain kinds of design flexibilities. In other words, at this stage there are certain steps you could take that would be much harder than when you had 80 percent construction, or 40 percent -- it depends on what you're doing.

MR. MYERS: Yes. To the extent they can define those today and say this is what's required, I agree with that.

MR. OKRENT: But you were laying it on the Staff to define this, in effect.

MR. EBERSOLE: As a case in point, you're a GSAR man. Browns Ferry has been generous in providing the industry with lots of flags as to what ought not to be done. First was the fire. More recently, we had the scram system fail. That was a handy signal to do something about the scram system.

Are you all doing anything about that with your GSARs, independently? Or are you waiting for the Staff to establish something to do?

MR. MYERS: Well, I'm not prepared to discuss the

details of that. I think if you look at the Browns Ferry scram problem, it was related to a piping situation --

MR. EBERSOLE: Yes.

MR. MYERS: -- but maybe --

MR. EBERSOLE: It was more than that, though.

MR. MYERS: -- Joe Quirk is here from GE and maybe he can address that specifically.

MR. EBERSOLE: Are there forthcoming some improvements beyond those which are presently on your GSARs?

MR. QUIRK: On the GSAR scram discharge volume --

MR. EBERSOLE: I'm aware of that.

MR. QUIRK: -- the particular configuration on the one at Browns Ferry --

MR. EBERSOLE: It was more than that, nowever.

MR. QUIRK: -- does not have the same --

MR. EBERSOLE: It does not have the same detailed specific malfunction potential. It has a related set, however, in that it looks at a dump volume which is closeable, which has a potential for filling --

MR. QUIRK: -- and which has instruments --

MR. EBERSOLE: Better instruments than Browns Ferry had; true.

MR. QUIRK: A different type.

MR. EBERSOLE: Right. But generically, though, it represents a problem area.

MR. QUIRK: It has been evaluated, yes. And we think that the BWR-6 scram system --

MR. EBERSOLE: You do not now intend to do anything on your own volition to GSAR BWR-6 in this area?

MR. QUIRK: I didn't say that.

MR. EBERSOLE: What?

MR. QUIRK: I did not say that.

MR. EBERSOLE: You didn't say that?

MR. QUIRK: No, I did not.

In response to the evaluation performed by the Staff, recommendations have been made by the Staff and by General Electric, as well; and that these recommendations will be carried out on our BWR plants.

MR. EBERSOLE: So you are doing something beyond the Browns Ferry catchup?

MR. QUIRK: Yes.

MR. EBERSOLE: Thank you.

MR. OKRENT: Mr. Lowenstein, before when I raised my question, Mr. Myers volunteered to comment. You indicated an interest in responding to my question.

MR. LOWENSTEIN: I'm not sure I remember anymore, Dr. Okrent, what that specific question was, but in response to your last question I would say that we have sent a letter to you, addressed to you, on behalf of Houston Lighting & Power indicating a desire to appear before the Committee in February

to discuss work which Houston has underway with its suppliers and its architect-engineers to look into the difficulties of implementing Option 3. They are considerable. The work has not been completed. We would expect that they will be brought more or less to at least an intermediate conclusion this month, and we would be in a position with representatives of the groups that are working with Houston to appear before you at that time and discuss in some detail the work that has been done and the conclusions that we have reached.

MR. OKRENT: But let me ask the Staff how the potential availability of the information just mentioned by Mr. Lowenstein impacts on your thinking and what you propose to do? Do you think you should wait to see what Houston Lighting & Power has developed before you arrive at a final recommendation to the Commission, or what?

MR. PURPLE: Well, I shared your earlier expressed maybe "concern" isn't the right word, but the comments that came from industry in response to the draft, didn't contain as much as we might have anticipated from earlier meetings with the subcommittee and our earlier understanding of what they had underway.

The Commission is pressing to move on with reaching a decision on this, and have asked us to bring down our best shot, which is what we're doing. It is hard to -- and we will do that. We will do that next Tuesday.

I don't know enough about what the industry is doing to really be able to ascertain how significant that should be, and should we hold off and wait until we see it, or proceed on with our own best judgment on the best information we have now. I really hadn't thought about this until you asked the question, but I think we would be likely to proceed on with what we have in this draft now and let that be our recommendation to the Commission next week.

MR. MATHIS: Well, Bob, let me toss something else out here.

MR. PURPLE: Sure.

MR. MATHIS: Option 1 basically says you go with things as they are now, plus the provisions that are going to be required from 0660. Right?

MR. PURPLE: Right.

MR. MATHIS: So there is a bounding element there that is pretty well defined.

MR. PURPLE: Right.

MR. MATHIS: And if you want to take advantage of a plant not being built and consider those major kinds of things that might effect, if you will, the fundamental design, if you go back to Jim's Option 3 which says: Okay, set aside some land over here. You can do that at this stage of the game, I am assuming. Provide some space under the unit, and beef up your structure. Those three things are pretty definitive.

And if you had that in addition to Option 1, would that make you happy? And could the utilities live with it? It seems to me you're covering your bets pretty well on anything that might come out of rulemaking on degraded core. I'm just tossing it out.

MR. MYERS: I think the problem with that, Mr. Mathis, is the Mark III's containment design indicates that the Skagit project -- I might say that it's a rather tight fit for everything that's in there now, and to decide you're going to jack something up a few feet, or whatever it takes to require space in there for it, is a reconsideration of the basic containment design.

We've learned a lot about the design of the Mark III containment, including loads and things that result from full swell, a lot of things that came into the process as we went through the proceeding. I suppose that we're really saying that once you do that, you decide you're going to change the design and allow a little space, which is easily said, we probably are going to reevaluate the basic design of the Mark III containment.

MR. MATHIS: All right, then let's --

MR. MYERS: I don't know how you can get away from that. That just means you're that much further away from being able to use it.

MR. MATHIS: Well, turn that coin over, though, and

say: All right, if degraded core rulemaking comes down the road -- and you're pretty well convinced, I think, that something is going to come down the road -- and you've been given your construction permit, and you're constructing, and then you get hit with that, aren't you in worse shape?

MR. MYERS: There are a lot of us in that shape, at that point, and if the judgment is that that's what needs to be done, I expect you may see the nuclear industry substantially reduced in the magnitude of the generation resulting from that resource.

MR. EBERSOLE: Well, isn't this --

MR. MATHIS: What I'm trying to get at is: Is there any way that you could get a more definitive bounding situation that would be compatible with what you have to live with, and satisfy the Staff? Because right now, we aren't together, I don't think.

MR. MYERS: I don't know of any way to provide for any eventuality in the design --

MR. MATHIS: Oh, I don't, either. I'm just trying to define one that we could live with.

MR. EBERSOLE: Well, I noticed the FNPs, though, didn't seem to be much concerned about this. We have just been told that the containment could be strengthened. Apparently you can put a crucible under it, or you can do several -- I was impressed.

MR. MATHIS: He's already done it.

MR. EBERSOLE: Impressed by the difference between the problem as described against the FNPs versus these other six stations. I suppose it has largely to do with the structural characteristics. It just happens that the FNPs are more adaptable, I presume. Your containment is harder to handle, I gather?

MR. MYERS: Well, you're drawing some conclusions that I guess I don't want to get into a discussion with you about, not knowing enough about all the different ramifications; but I think it is fair to say that one of the advantages of the OPS system is there aren't any they have to backfit.

MR. EBERSOLE: Yes, that's right.

MR. MYERS: That's fairly clear. I don't know how many they've got under construction --

MR. EBERSOLE: None.

MR. MYERS: -- but I think it is a fairly limited number. So I don't know that we could really mix those two into the bowl --

MR. EBERSOLE: Yes, you're further down the road, I think.

MR. LOWENSTEIN: I think this is one of the questions -- Excuse me. Robert Lowenstein -- which is being looked at in the Houston study. My expectation is that we can talk further about it at that point.

Could I add something, a comment really going back to Dr. Okrent's question to me earlier. One of the things we are looking for here is -- and this has been an underlying subject of discussion -- is some fairly clearcut criteria which will let these utilities go ahead into a hearing with the kinds of questions we're discussing now being put off into rulemaking proceedings, the effort being that these kinds of basic redesign issues not be taken up case-by-case. I think it is perfectly evident from the discussion this morning that the Staff doesn't have positions on the particular items; that what there are are a series of spectrum, if you will, of uncertainties which would then be resolved not between now and the Commission meeting next week, but case by case, from one PWR or one BWR to the next. Well, we know from the history of these particular applications, when there were fewer licensing uncertainties than there are today, that you can't get there from here and get a permit, or a license issuance, given this magnitude of uncertainty.

I think what will come out of the discussion in February is that some of the costs of making provision for these various possible mitigating or preventive systems, without knowing exactly which ones you will need to in fact install later -- because we can't know before the rulemaking proceeding is over -- involve a great deal of cost, involve a great deal of re-engineering, and throw into doubt much of the

sunk cost which the utilities have already made, and which they are trying to rescue now.

You had asked earlier whether there were any legal insights, Dr. Okrent. I haven't heard anyone yet say that the existing criteria which have been approved for reactors generally are not adequate for these plants. I have heard identification of many issues and questions that it might be desirable to consider, and that it might save money for the utilities to consider now, but I haven't heard anyone say that they're inadequate for the issuance of a construction permit, or less adequate for this group of plants than for the great bulk of plants that are under construction or in operation.

I think it is perfectly evident that there is no way that this Commission, or this Congress can commit future Commissions or future Congresses to some kind of grandfathering provision. These utilities will be subject, in one way or another, to the outcome of the degraded core rulemaking proceeding. What they are basically asking for is the opportunity to go ahead with their permits, not to reopen all the myriad of questions that could be asked, but to be subject to new requirements as they come along and as they are adopted by the Commission for other reactors in the same class.

MR. OKRENT: One of the questions that I had Dr. Savio send out in an effort to focus, or stimulate discussion was what safety goals the utilities would suggest or

endorse. You a moment ago used the word "adequate." So I guess in a sense that's what's on the table. Were the previous criteria adequate? I guess in general one has to assume that the NRC has found not, because they have found a lot of things to require, since TMI, and there are suggestions -- otherwise they wouldn't at least be considering these rulemaking hearings -- that there may be other things.

I would appreciate it if you would tell me what is either your or your client's definition of "adequate," and why.

MR. LOWENSTEIN: I can't be precise for any particular client, now, but I think I can give a generalized answer. That is, the Option 1 would provide an adequate basis, in the sense that the Commission and the ACRS, with the post-TMI lessons learned, and the other post-TMI requirements, had found that it is safe for these plants to operate on that basis.

Now that does not mean to say that there are no further improvements which can be made. Of course, there can always be. We have heard a discussion today of many different kinds of improvements, and different people have differing attitudes as to what should be looked at first and considered initially. But I think it goes without saying, because the plants are operating, that this committee and the Commission have licensed additional plants for operation since TMI, and

will be licensing further plants, and that that is adequate to protect the public health and safety without foreclosing the possibility that additional improvements can be made.

I think what we are urging here is that that same basis is adequate for the NTCPs, and that the NTCPs would recognize that further requirements which may be adopted by the Committee and the Commission and the Staff from time to time would be imposed on them as well; but that they not be held up until these other decisions are made.

MR. OKRENT: Well --

MR. LOWENSTEIN: Now this isn't the objective answer, but it is a general statement and I think a pragmatic approach to getting on with this --

MR. OKRENT: I have a little bit of a problem philosophically and really also from a practical point of view with your use of the term "adequate." I will use an example from Los Angeles where I live.

For some years now there has been a discussion in the City Council and in the newspapers about a large number of buildings, masonry buildings, not designed for seismic effects pre-1933 designs, multiple dwellings or large commercial buildings, and so forth. If I took the way you just used "adequate," since they were allowed to be occupied last year, and in fact are allowed to be occupied today, they must be "adequate," because otherwise how could the City Council permit

this to go on?

Nevertheless, the City Council has been struggling for some years now -- and believe me, it is a struggle -- with possible ordinances which might require backfitting. The cost of the backfitting has ranged, but it goes up to a billion dollars, an order of magnitude, and some people say we will have to tear the building down and make people move from low-rent to high-rent areas. You have lots of complicating problems.

Nevertheless, I would have a problem assuming it was adequate just because people had been allowed to live there. And certainly the City Council wouldn't allow a new building built tomorrow next door to the existing building to be built the same way. In fact, they haven't allowed it for some years.

MR. LOWENSTEIN: I don't think the analogy fits, though, with all due respect.

MR. OKRENT: I'm just saying that one has to be a little cautious. I deliberately chose an example where neither you nor I would not be able to say the existing buildings were adequate, but nevertheless they have been allowed to persist.

MR. LOWENSTEIN: Because I know that this Committee has reviewed the situation for reactors since TMI. We're both familiar with the changes that have been made, and

the progress that has been made.

I think where we're having trouble now is that Option 3, for example, says we don't know what further requirements might be necessary, and we're not even telling you which alternatives potentially to make provision for. There is no decision-making being made; it's just a sort of generalized wringing of the hands, if you will, that some day there may be some future requirements that we may want you to put into effect. We're not prepared to tell you which are likely or unlikely, and in fact we're not prepared to define all of them for you.

And under these circumstances, it really isn't very practical for you to come back even to this Committee for a report on its PSAR, or to amend its PSAR to get through a litigated hearing. And every one of these applications has intervenors, and in some cases many of them, which is why they have been open and pending for so long -- one of the reasons.

MR. OKRENT: Well, I am quite conscious of the kind of problems you have just been discussing. That is why I am trying to carry the discussion out --

MR. LOWENSTEIN: But I do want to be clear. We don't want to be in the position of the houses that people are living in, because you have no place else to put them. We are all conscious of the extensive post-TMI retrofitting and the further work that is going on on these plants. When I say

"adequate," I mean "safe."

MR. OKRENT: Oh, please. I have a thing about -- What does it mean when you say something is "safe"? That is a word I am unable to define. It is used in so many different contexts and ways. I will give you an example.

I read in the Los Angeles Times -- my fountainhead of information -- that Secretary of the Interior Andrus had said that the proposed Auburn Dam was safe. Those were the words that the LA Times used. I assume that he used them.

Now if you don't know the history of this Dam, it is above Sacramento, or would be above Sacramento. It has been estimated that, were it to be built, and filled, and were it to fail, it could lead to three-quarters of a million fatalities.

All right, now what does the term "safe" mean in the context of something that poses that hazard? Because I am sure there is not a zero probability of failure. Nevertheless, the term "safe" is used.

You can see why I prefer that we don't use it here.

Well, there appears to be a little problem.

(Laughter.)

MR. LOWENSTEIN: I would hope the Staff would communicate to the Commission our plan to come back and report to you further on some of the engineering difficulties at the February meeting.

MR. OKRENT: I notice that each time you have

mentioned what it was you planned to report in February, if there were such a meeting, it would be on the engineering difficulties. Were there going to be some proposals for possible changes beyond the previous design? Or was it just on the engineering difficulties?

MR. LOWENSTEIN: My colleague, Dave Powell, David Powell, has been involved really very extensively with this work.

MR. POWELL: Dr. Okrent, what Houston has done is to take the NTCP rule as proposed, the degraded core rulemaking section, and has -- Vasco, Saul Levi, Incorporated (phonetic), and General Electric -- looking at that proposed rule and undertaking engineering studies.

The rule as proposed says: Look at mitigative features from the standpoint of whether they're practical given the state of the art of the technology at this time, and determine mitigative features which you could commit, not to foreclose during construction.

Also, look at preventive measures -- implicit in that being: We hope that there is some relationship between preventive measures and mitigative measures as far as risk reduction is concerned. And come back to the Commission with a report on what you will commit not to foreclose, and give us a plan on how you will implement that.

Now these organizations are looking at various kinds

of mitigative measures, those that are specifically listed in the proposed rule, and others, from the standpoint of determining whether, for the basic design those can be reasonably accommodated insofar as they have committed not to foreclose. Or, if they cannot be reasonably accommodated, what are the reasons that they cannot be reasonably accommodated.

They are also looking at various kinds of preventive measures from the standpoint of how one would actually incorporate those in a design in the sense that the rule speaks of. What is the potential risk reduction of each various preventive measure.

And then looking at all of this together, what is the best combination one could put together that would give you the potential for an overall reduction of risk from that plant?

So it is a rather extensive effort, done on a very rushed basis, because the Islands Creek plant is looking towards health and safety hearings later this year in the spring. But I believe it is a good-faith effort, and it is not an effort designed only to give the litany of reasons why you cannot do things; it's really an effort to see what combination of things might reasonably be provided for. And it is an attempt to put some meat on whatever one views as the vagueness of the proposed rule.

And in that sense, it seems to me to be a rather useful effort.

MR. OKRENT: Thank you.

MR. POWELL: And we would hope that before the Commission acts, they would take the benefit of that.

MR. OKRENT: Yes. It was my impression of what was being done was rather like what you had described, and I didn't really want it left as more negative, or however you wanted to phrase it. So thank you for the elaboration.

I wonder, is there a similar study being done for a PWR, other than Zion and Indian Point?

(No response.)

MR. OKRENT: I don't hear any volunteers.

Well, are there other points the subcommittee would want to raise now?

Let's see. With regard to the meeting with the Committee, I guess that we are scheduled for 8:30 to 11:30, and I doubt that will be allowed to run late since Sequoyah is on next after this.

I would suggest you assume that we may begin the discussion with the NRC Staff and the industry before 9:30. I see that the schedule shows an hour-long subcommittee report, but I doubt very much that that will be the case. I suggest you be ready by 8:45.

(Laughter.)

MR. OKRENT: And I think, by all means, bring in something in writing for the areas of reliability engineering,

and degraded core, that the committee can look at.

MR. PURPLE: Yes.

MR. OKRENT: And I think we should hear the discussion on siting, and maybe it would be just as well to start with it as we did -- let me think a minute.

MR. MULLER: Do you want to go through this comparison? It's sort of a red herring.

(Pause.)

MR. OKRENT: All right, let's make it third instead of first of the trilogy. Okay? So in other words, reliability engineering; degraded core; and then siting. But I think we should have them here, for what you have to say and also what your thinking is, the directions in which you are going. That's the most interesting part. Okay?

MR. MULLER: Yes. But do you still want the comparison with the NUREG-0625?

MR. OKRENT: Yes, I think pretty much the same material you covered, but let's, in other words, have it third in the group. I suspect the changes in 0718 you can mention and see if the committee wants to pick up a specific question. We'll just have to see. But I would prefer not opening up a long list of 40 things, at least not at the beginning of the three hours, or we'll never get to part two.

And let's see. Who will we have, if anyone, from the NTCP owners' group on Friday? Do we know?

MR. LOWENSTEIN: Mr. Myers has just -- he and Mr. Mecca have just left for Seattle and will not be able to be back.

MR. OKRENT: Will someone be here from Houston Power & Light?

MR. LOWENSTEIN: (Nodding in the negative.)

MR. OKRENT: No? Okay -- D. Walker? Will you be here?

MR. WALKER: I plan to be here, or Mr. Hagan, one or the two of us will be here.

MR. OKRENT: One of the two of you will be here from FNP. But at least there is no one requesting time to make a statement, and we may not have a representative from the NTCP group? I am somewhat surprised --

MR. LOWENSTEIN: Let us consult and we'll get back to you.

MR. OKRENT: Yes. It needs --

MR. LOWENSTEIN: We weren't aware of the Friday meeting, really, until just the last minute.

MR. OKRENT: Oh, I see.

MR. LOWENSTEIN: Just the day before yesterday. That's part of the problem.

MR. OKRENT: I think, myself, it makes sense for there to be someone from the NTCP group.

MR. LOWENSTEIN: We'll do our best to have someone

here. We appreciate the opportunity to do that.

MR. OKRENT: At least to offer informal, off-the-cuff comments. It need not be a formal presentation.

Okay, I would like to recommend that you set up your presentations so that if they were uninterrupted, you know, they would -- oh, I suspect Mr. Myer needs ample time, because he has a fairly long thing to show -- but I will encourage the Staff to put forward its best effort, trying to again indicate the pros and cons of the alternatives as they see them and, to the extent you can, indicate the problems that the NTCP owners see, and that you see, or whatever, in what you do, and so much the better. Okay?

MR. PURPLE: Okay.

MR. OKRENT: In other words, there is little advantage in your giving a very short presentation that is incomplete, and so forth, and then having a disorganized question and answer period that doesn't make the point except with difficulty.

I think since the subcommittee report is going to be short, you should have ample time to make a good presentation.

MR. PURPLE: There's about two hours and 45 minutes, perhaps, without interruption. Should we plan on an hour-and-half?

MR. OKRENT: You should figure half of that. Yes, or a little less, I would say. Okay? But that should give you

time to do a good presentation. Cut out the history, and this sort of stuff. Get to the hard part early.

MR. PURPLE: Okay.

MR. OKRENT: Anything else?

(No response.)

MR. OKRENT: All right, then I will adjourn the meeting.

(Whereupon, at 4:00 p.m., the meeting was adjourned.)

* * *

end
JWB

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

ACRS Subcommittee on Safety Philosophy

in the matter of:

Date of Proceeding: January 6, 1981

Docket Numbe :

Place of Proceeding: Washington, D. C.

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

Mary C. Simons

Official Reporter (Typed)

Mary C Simons

Official Reporter (Signature)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

ACRS Subcommittee on Safety Philosophy

in the matter of:

Date of Proceeding: Jan. 6, 1981

Docket Number:

Place of Proceeding: Washington, D.C.

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

Ann Riley

Official Reporter (Typed)

Ann Riley

Official Reporter (Signature)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

ACRS Subcommittee on Safety Philosophy, Technology and Criteria

in the matter of:

Date of Proceeding: Janaury 6, 1981

Docket Number:

Place of Proceeding: Washington, D. C.

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

Jane N. Beach

Official Reporter (Typed)

Jane N. Beach

Official Reporter (Signature)

Schwencer

The NRC requested comments on proposed licensing requirements for pending construction permits and manufacturing license by a notice in the Federal Register on October 2, 1980. In response, comments were received from:

C. W. Rowley, Sand Springs, Oklahoma (Rowley)

Department of the Interior, Office of the Secretary (USDIOS)

Marvin I. Lewis, Philadelphia, Pennsylvania (Lewis)

Bechtel Power Corporation, San Francisco, California (Bechtel)

Lowenstein, Newman, Reis, Axelrad & Toll (Lowenstein, et. al.)

Offshore Power Systems (OPS)

Public Service Company of Oklahoma (PSO)

Boston Edison Company (BEC)

General Electric Company (GE)

Westinghouse Electric Corporation (W)

Portland General Electric Company (PGE)

Duke Power Company (Duke)

ENCLOSURE 3

CHANGES IN CATEGORY ASSIGNMENTS

NUREG-0660 ACTION ITEM		PRELIMINARY ASSIGNMENT FOR CP/FOR ML	REASSIGNMENT CP/ML
I.B.1.1	Organization and Management Long Term Improvements	5/5	2/1
I.C.1	Short Term Accident Analysis and Procedures Revision	4/4	2/1
I.C.5	Procedures for Feedback of Operating Experience	4/4	5/5
I.D.4	Control Room Design Standard	4/4	1/1
I.F.1	Expand QA List	4/4	5/5
I.F.2	Develop More Detailed QA Criteria	4/4	5/5
II.A.2	Site Evaluation of Existing Facilities	5/1	4/1
II.D.2	Research on Relief and Safety Valve Test Requirements	3/3	1/1
II.E.2.1	Reliance on ECCS	2/2	2/1
II.E.2.3	Uncertainties in Performance Predictions	3/3	1/1
II.K.1.20	Prompt Manual Reactor Trip Procedures and Training	4/1	2/1
II.K.1.21	Safety Grade Anticipatory Automatic Reactor Trips	4/1	1/1
II.K.2.2	Procedures and Training to Indicate and Control AFW independent of ICS	4/1	1/1
II.K.2.13	Effect of HPI on Vessel Integrity for Small-Break LOCA w/o AFW	3/1	1/1

ENCLOSURE 3

CHANGES IN CATEGORY ASSIGNMENTS

(CONTINUED)

NUREG-0660 ACTION ITEM		PRELIMINARY ASSIGNMENT FOR CP/FOR ML	REASSIGNMENT CP/ML
II.K.2.15	Effects of Slug Flow on B&W Steam Generators After Primary System Voiding	3/1	1/1
II.K.3.3	Report Safety and Relief Valve features and challanges	2/2	2/1
II.K.3.5	Continue to Study Need for Automatic RCP Trips and Need for C.1.4.c, etc.	4/4	1/1
II.K.3.30	Revise Small-Break LOCA Methods	3/1	1/1
II.K.3.31	Do Plant Specific Calculations re: 10 CFR 50.46	3/1	1/1
II.K.3.46	Respond to ACRS Consultant Concerns	4/1	1/1
III.A.1.1	Upgrade Emergency Preparedness	5/1	1/1
III.A.2.1	Amend 10 CFR 50 and Appendix E	5/1	4/4
III.A.2.2	Development of Guidance and Criteria	5/5	1/1
III.D.1.2	Radioactive Gas Management	4/4	1/1
III.D.1.3	Ventilation System and Radioiodine Adsorber Criteria	4/4	1/1
III.D.2.3	Liquid Pathway Radiological Control	2/2	2/1

REVISED TEXT OF REQUIREMENT

(SEE APPENDIX C TO PROPOSED FINAL DRAFT OF NUREG-0718)

I.C.5 - EXPANDED FOR BETTER GUIDANCE

I.D.1 - STRENGTHENED REQUIREMENT

I.D.2 - ADDED REFERENCE TO NUREG-0625

I.F.1 - STRENGTHENED REQUIREMENT

II.A.2 - REVISED - PROPOSED RULE HAS SLIPPED

II.D.1 - STRENGTHENED REQUIREMENT

II.E.4.2 - REVISED AND EXPANDED FOR BETTER GUIDANCE

II.F.2 - ADDED REFERENCE TO NUREG-0737

II.F.3 - REVISED TO REFLECT SLIP IN ISSUANCE OF R.G.1.97

II.J.3.1 - STRENGTHENED REQUIREMENT

II.K.3.13 - MINOR CLARIFICATION

II.K.3.28 - STRENGTHENED REQUIREMENT

III.A.2.1 - UPDATED TO REFLECT ISSUANCE OF FINAL RULE

DUKE POWER COMPANY
GENERAL OFFICES
422 SOUTH CHURCH STREET
CHARLOTTE, N. C. 28242

TELEPHONE: AREA 704
373-4011

January 5, 1981

Mr. David Okrent, Chairman
ACRS Subcommittee on Safety Philosophy,
Technology and Criteria
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

On October 2, 1980 the NRC published a Notice of Proposed Rulemaking concerning the "Proposed Licensing Requirements for Pending Construction Permits and Manufacturing License Applications". This proposed rulemaking is being developed to consider the requirements which must be met in the design of nuclear power plants which are pending construction permit approval. Duke Power Company currently has three construction permit applications (Perkins Nuclear Station) pending before the Commission and, as such, is deeply concerned with the proposed rule.

Duke is well aware that the Commission has, based upon its extensive review of the Three Mile accident, approved an action plan which presents a sequence of actions intended to result in the increasing improvement of safety in the design and construction of nuclear power plants. The Commission is now developing its position with regard to near term construction permits. As stated in the proposed rule, there are three options which have been considered. These options have different impacts upon the various utilities which must now make the vital decision as to whether the construction permits applications which are now pending will be carried to fruition.

We have a very difficult time in separating the Cherokee Nuclear Station units, which were approved for construction in December, 1977, and the Perkins Nuclear Station units, which are still without a CP. Because of their standardized design, construction permit applications for these six units were filed simultaneously in March of 1974; in fact all six of these units shared one common PSAR. All six units were intended to be identical, and it is our objective to continue the standardized concept in these units. By applying different requirements to the three units at Cherokee compared to the three units at Perkins the standardized concept will

not be achieved. We believe that the Perkins units should be exempted from this rulemaking on this basis and that construction permits should be quickly issued for those three units.

In its action plan the Commission has identified four areas that it believes merit special attention. The first area to be considered is siting. We regard the siting and the questions concerning the siting of the Perkins Nuclear Station to be answered. The Atomic Safety and Licensing Board has issued its partial initial decision in the matter. The selected site and the alternative sites were considered. The Perkins and Cherokee units were sited at the same time, in the same method, and with the same degree of certainty. We, at Duke, cannot understand why additional siting requirements should be imposed at this time. Any commitment on our part to meet whatever requirements ultimately come out of the Advance Notice of Proposed Rulemaking on power plant siting will be a commitment to start the siting process anew when that rule is adopted. It is significant to note that the NRC Fiscal Year 1980 authorization bill funds the establishment of demographic requirements, yet specifically excludes any facility for which a construction permit application was filed on or before October 1, 1979. As mentioned before the Perkins application was filed in March, 1974--5 1/2 years before the bill's cutoff date.

The degraded core rulemaking will require applicants to indicate their degree of conformance to the interim rule. We think Perkins should be exempted from the rulemaking since the Cherokee and Perkins stations consist of identical standardized units. Whatever changes are finally made at Cherokee will then be made of Perkins. Otherwise, the standardization will be lost.

The area of reliability engineering and the area of emergency preparedness can both be adequately handled after the issuance of construction permits on the Perkins units. Standardization will affect the reliability of the Perkins units just much as it does the Cherokee units. As stated before, Duke intends to make the Cherokee and Perkins identical and would build them to the same degree of reliability. The emergency preparedness questions will have to be handled on a site by site basis. Based upon our review of the emergency preparedness rule, (Appendix E to 10CFR 50) we have already provided

enough information to indicate with reasonable assurance that emergency preparedness requirements can be implemented at the Perkins site.

We find it very difficult to believe that any applicant would continue its quest for a construction permit in the face of the proposed rule which has been issued in the Federal Register. Those utilities with pending applications must now make the decision as to whether construction can and will start at their proposed site in the face of the commitments which must be made in order to obtain that construction permit.

We do believe that construction permits can be issued for the 11 plants which are now pending before the Commission without the adoption of the proposed rule. To do this the Commission would have to take the strong step of convincing itself that the current rulemakings underway can be handled by the applicants concerned and that it is in the best interest of the nation's energy future to begin the construction of those plants.

Duke urges you, the ACRS, to support Option 1 of the proposed rule as the best and most logical approach to continuance of the nuclear power-generating option.

Very truly yours,

J E Beall / for

L. C. Dail, Vice-President
Design Engineering

Meyer

LIST OF POTENTIAL DEGRADED CORE MITIGATION FEATURES

1. Energy Removal thru Containment Heat Removal (Passive or Active) -- no Radiological Release to Atmosphere.

2. Energy Removal thru Containment-Atmosphere Mass Removal (Filtered, Vented, Containment Systems) -- Controlled, Small Radiological Release.

3. Energency Dilution thru Increased Containment Volume (Venting to Large-Volume, Leak-tight Structure) -- No Radiological Release to Atmosphere.

4. Energy-Release Suppression thru Suppression of the Burning of Combustibles (Hydrogen and Other Combustibles) Control thru Suppression of Buring, e.g., Inerting, Halons, Water Mists) -- No Radiological Release to Atmosphere.

5. Energy-Release Management thru Controlled Burning of Hydrogen and Other Combustibles (Hydrogen Ignition Systems) -- No Radiological Release to Atmosphere.

6. Energy-Release Control and Core Mass Management thru Core Retention Devices (Core Catchers, Core Ladle, Cavity Flooding) (Active and Passive Cooling)
   -- No Radiological Release to Atmosphere
   -- No Impact or Considerably-Reduced Impact to Liquid Pathway.

7. Kinetic Energency Dissipation of Steam-Explosion-Generated Missiles thru Missile Shields -- No Radiological Release to Atmosphere.

8. Energy Absorption Enhancement thru the Strengthing of Containment Structures

Put in at last pg

Portland General Electric Company

Bart D Withers Vice President

January 2, 1981

Pebble Springs Nuclear Plant
Docket 50-514, 50-515
License NPF-1

Dr. David Okrent, Chairman
Safety Philosophy, Technology
and Criteria Subcommittee
Advisory Committee on Reactor
Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Okrent:

Portland General Electric Company received a telecopy of your request for the near-term Construction Permit (CP) applicants to address the NRC Staff recommendations for resumption of CP licensing on December 18, 1980. Consistent with the current level of inactivity on our Pebble Springs CP application brought about by incessant licensing delays, we are unable to attend the subcommittee meeting scheduled for January 6, 1981. However, we would like to offer the following comments for the subcommittee's consideration.

We have submitted comments on the NRC's Notice of Proposed Rulemaking regarding proposed licensing requirements for pending CP applications (45 FR 65247). We believe that the NRC Staff's approach for licensing pending CP applications as proposed therein is unwarranted and unworkable. As an alternative, we strongly recommend that the Commission adopt a policy that would precondition CP licensing on meeting pre-TMI requirements, as augmented by applicable and currently defined requirements flowing from the TMI-2 accident (ie, Option [illegible]. This appears to us to be the only feasible way to proceed with licensing of pending CP applications.

We regard the use of quantitative safety goals in determining the need for design changes as a noble pursuit. Indeed, we agree with the ACRS recommendation, as stated in NUREG-0739, that it would be desirable to perform an analysis to form a safety profile of a particular plant and site that could be used to make risk-based decisions on design and/or procedural changes. However, the framework envisioned and espoused in NUREG-0739 for probabilistic risk assessments would threaten the viability of pending CP applications since risk quantification and certification will take years to develop into a disciplined and predictable process. From a corporate risk standpoint, pending CP applications can ill afford to be treated as prototypes for improvising a licensing process based on risk-acceptance criteria.

121 S.W. Salmon Street, Portland, Oregon 97204

The Pebble Springs location is one of the best in the contiguous United States for siting a nuclear power plant because of favorable demographic, seismic and evacuative characteristics. We also believe that the Pebble Springs nuclear and balance of plant design, as augmented by applicable lessons learned from TMI-2, is fundamentally sound and an improvement over currently operating reactors and those under construction. Notable plant design features which decrease vulnerability to serious accidents include a large dry Containment, a four-train Auxiliary Feedwater System independent of all AC power, and a steam generator layout conducive to natural circulation. Moreover, overall plant design more generally conforms with Standard Review Plan acceptance criteria, Branch Technical Positions and Division 1 Regulatory Guides. We therefore believe that additional design changes of the type intimated in your questionnaire are not warranted to decrease the vulnerability of our plant to serious accidents. What we consider to be a highly acceptable plant safety design is further accentuated by the demonstrably low population densities surrounding the Pebble Springs site and favorable site evacuation characteristics.

Sincerely,

B.D. Withers

Bart D. Withers
Vice President
Nuclear

c: Mr. Lynn Frank, Director
State of Oregon
Department of Energy

Myers

Figure 1

SERVICE AREA MAP

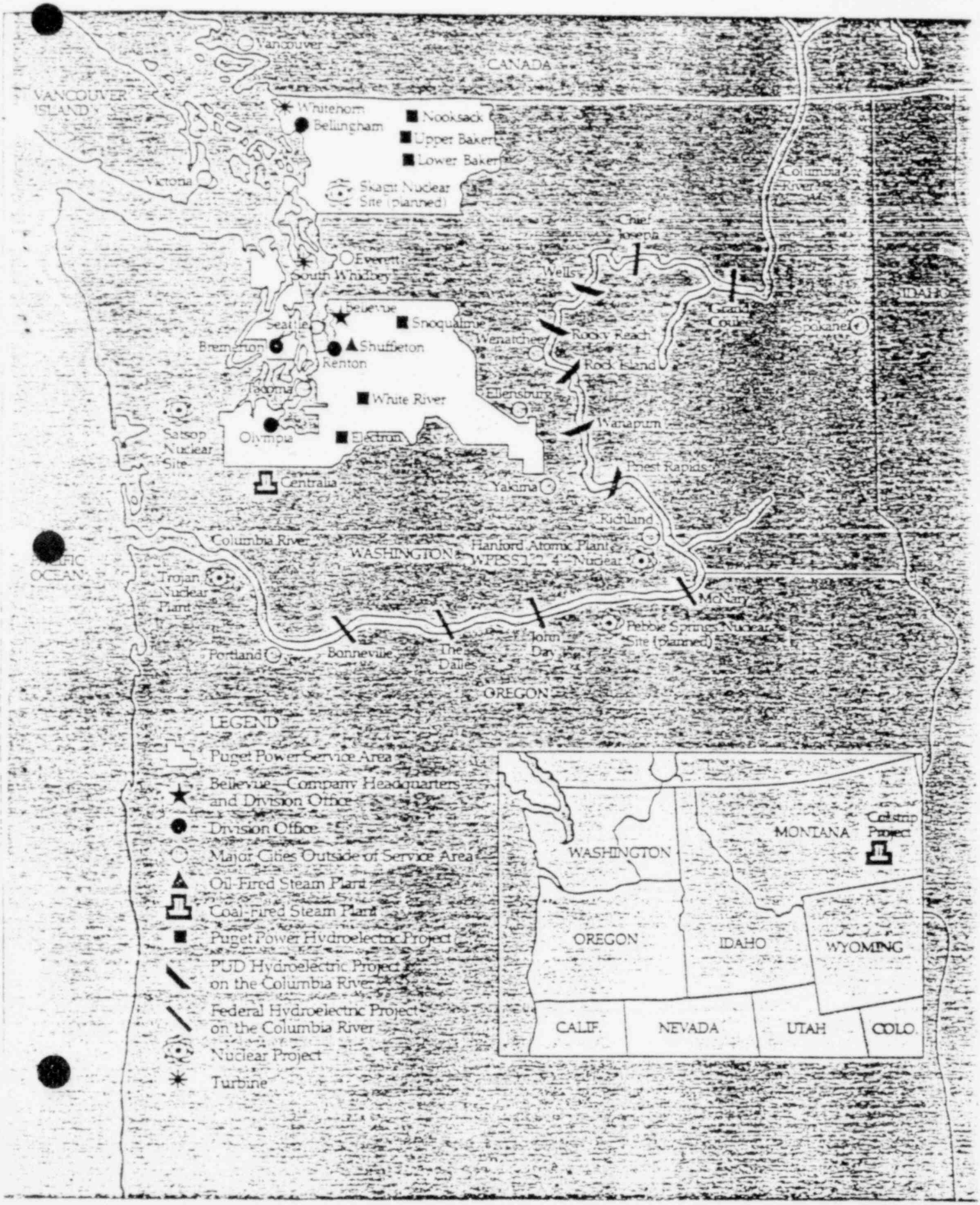
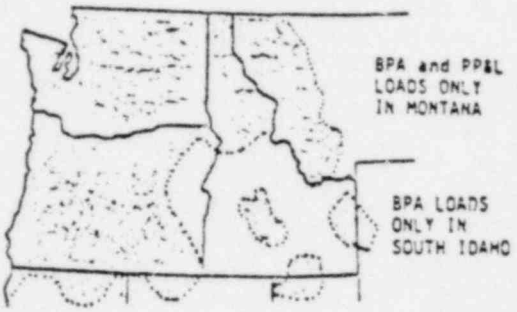


Figure 2

Northwest Power Pool

ENERGY LOADS AND RESOURCES
(Adverse Hydro)



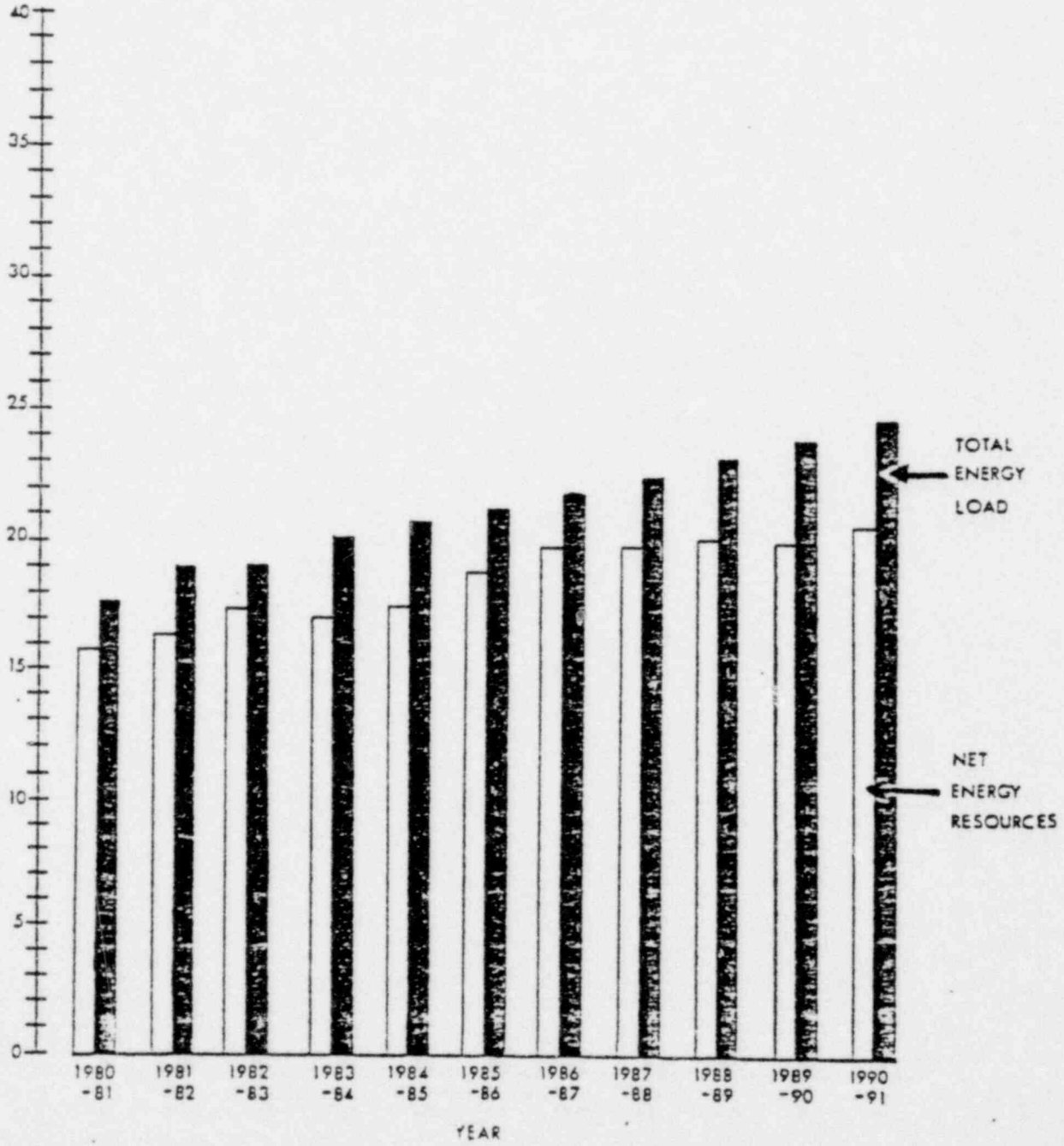
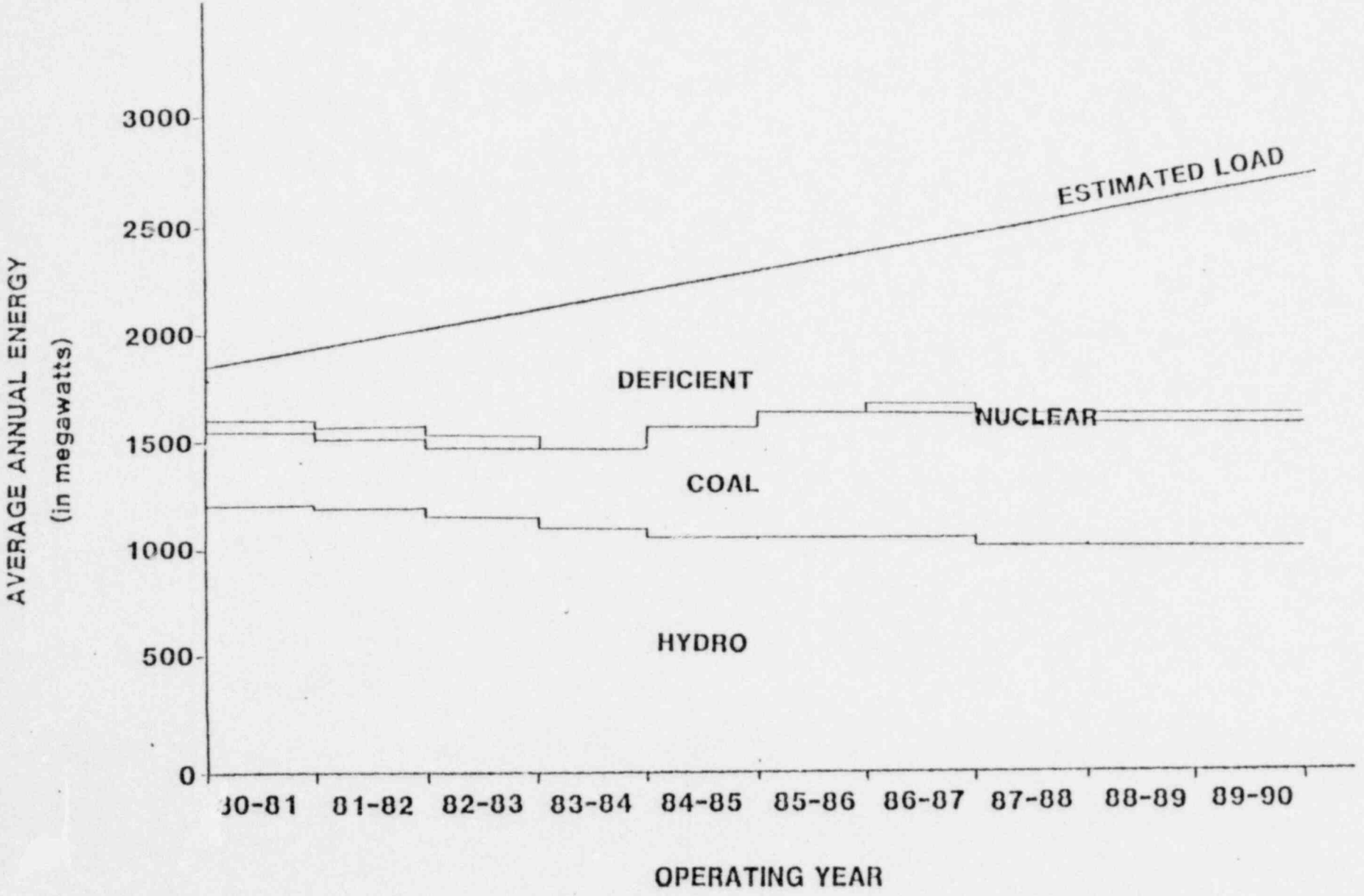
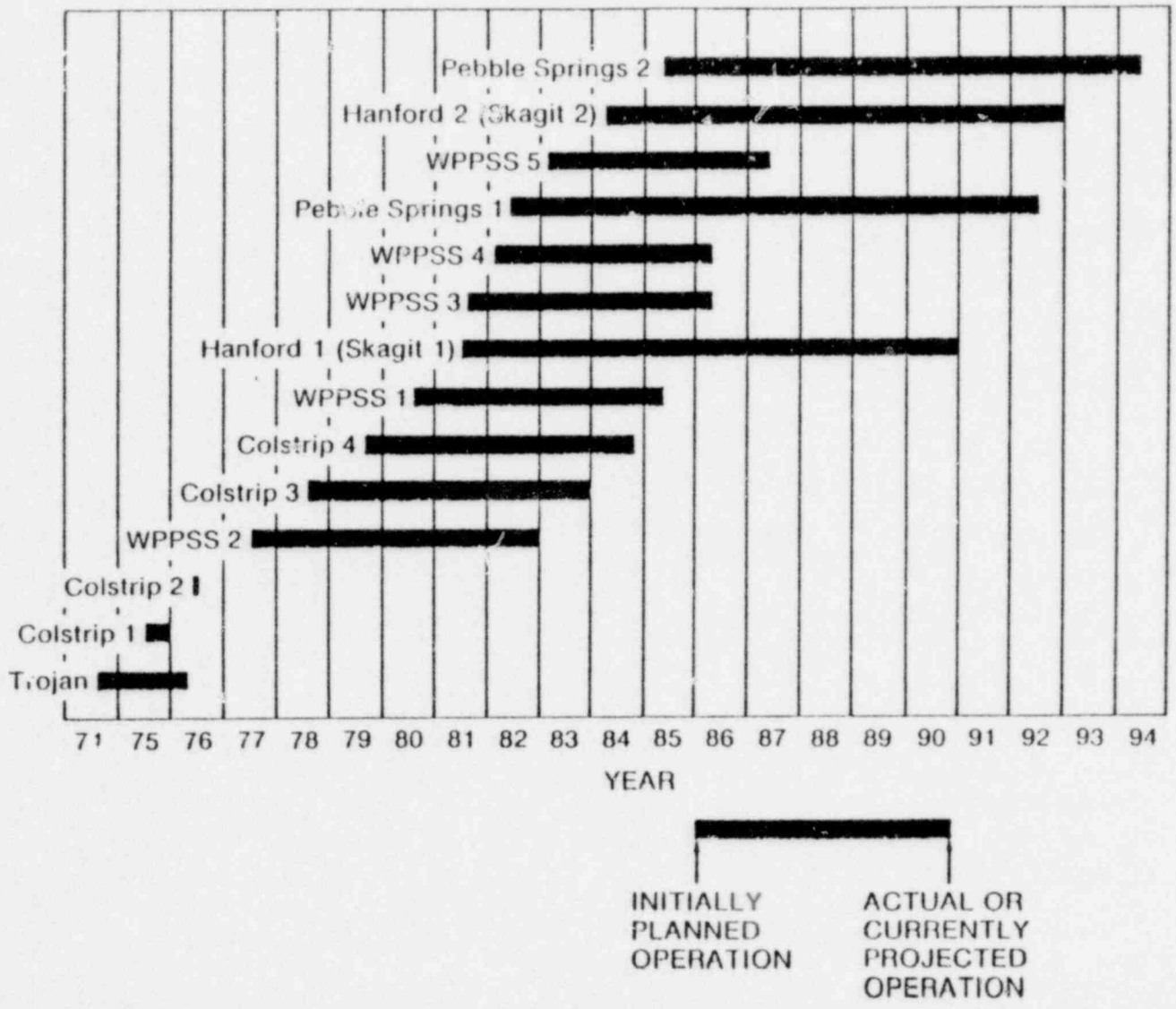


Figure 3

PUGET SOUND POWER & LIGHT CO.

LOADS AND RESOURCES





Thermal Plant Schedule Slippages

COMPARISON OF NTCP SITES WITH
ILLUSTRATIVE CRITERIA IN NUREG-0625

SITES	0-5 MI	MAX SECTOR TOTAL	5-10 MI	MAX SECTOR TOTAL	10-20 MI	MAX SECTOR TOTAL
NUREG-0625	100/MI^2 (0.5 RPD*)	3,925	150/MI^2 (0.75 RPD)	17,670	400/MI^2 (2 RPD)	188,500
ALLENS CREEK	23	1,028	26	2,685	28	7,707
BLACK FOX	45	1,312	28	1,193	127	67,732
PEBBLE SPRINGS	5	375	2	312	0.3	310
PERKINS	57	2,252	125	4,637	187	39,805
PILGRIM-MA-POP=727	122 (363)	5,342 (14,247)	84	7,365	179	30,753
SKAGIT	48	1,415	52	7,743	34	12,198

*RPD-REGIONAL POPULATION DENSITY
REGION IN THIS ANALYSIS IS STATE

SITING

POLICY ON PROCEEDING WITH PENDING

CP AND ML APPLICATIONS

1. CP'S AND ML'S MAY BE ISSUED FOR THOSE APPLICATIONS THAT MEET THE REQUIREMENTS OF OUR CURRENT SITING REGULATION, 10 CFR PART 100, AND HAVE BEEN EVALUATED AS SPECIFIED IN REGULATORY GUIDE 4.7.

2. AFTER PROPOSED SITING RULE IS ISSUED FOR COMMENT, BUT NOT AS A CONDITION FOR A CP OR ML, APPLICANTS WILL REVIEW THEIR SITES AGAINST THE NEW RULE.

3. STAFF WILL EVALUATE SITE CHARACTERISTICS AND PLANT DESIGN AND DECIDE WHETHER DESIGN MODIFICATIONS ARE RECOMMENDED.

HOUSTON LIGHTING & POWER COMPANY
P. O. Box 1700
Houston, Texas 77001

January 2, 1981

Dr. David Okrent, Chairman
ACRS Subcommittee on Safety Philosophy,
Technology and Criteria
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Okrent:

This will confirm the conversation which Robert Lowenstein had with you regarding the planned meeting on January 6, 1981, of the ACRS Subcommittee on Safety Philosophy, Technology, and Criteria, and our suggestion that the meeting either be rescheduled for the first week of February or another session be held at that time to receive our views.

Since our Allens Creek project would be subject to the proposed Near Term Construction Permit rule, we are concerned that the final rule provide adequate guidance for NRC Staff and Atomic Safety and Licensing Board consideration of construction permit applications and the necessary licensing basis for issuance of construction permits. We view the subcommittee meeting as an opportunity to discuss these concerns.

We are unable to attend the January 6 meeting, however, for two reasons. First, I and others in our senior management are already scheduled for other important meetings that day which cannot be changed. Second, the studies which we have underway based on the proposed NTCP rule are not yet at a stage which is suitable for meaningful discussion with you.

By the first week in February we will be able to convey to you our concerns with the NTCP rule as proposed and to furnish the ACRS with a presentation as to the studies which we have underway. This presentation will identify the scope of the studies concerning preventive and mitigative measures, the potential for risk reduction, and, depending on the status of the results at that time, where the studies might lead us.

I suggest February 3, 5 or 6 as suitable dates, and look forward to discussing this important matter with your subcommittee.

Sincerely,

George Oprea, Jr.
Executive Vice President

BOSTON EDISON COMPANY
800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

J. EDWARD HOWARD
VICE PRESIDENT
NUCLEAR

January 2, 1981

Dr. David Okrent, Chairman
Subcommittee on Safety, Philosophy,
Technology, & Criteria
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Proposed Licensing Requirements for Pending Construction Permit Applications

References:
(1) Notice of Proposed Rulemaking, same subject (45 Fed. Reg. 65247, October 2, 1980)
(2) Letter: J. E. Howard (BECo) to the Secretary (NRC); same subject, dated: November 17, 1980
(3) Advance Notice of Proposed Rulemaking,Consideration of Degraded or Melted Cores in Safety Regulation (45 Fed. Reg. 65474, October 2, 1980)
(4) Letter: D. C. Gibbs (AIF) to Secretary (NRC), Consideration of Degraded Cores in Safety Regulation, dated: December 31, 1980

Dear Dr. Okrent:

By reference (1), the Nuclear Regulatory Commission on October 2 published and sought comment on a "notice of proposed rulemaking"; this notice delineated the Commission's plan to take into account lessons learned in connection with the Three Mile Island Unit 2 (TMI) accident for nuclear power plants which are the subject of pending construction permit applications. By reference (2) Boston Edison, on November 17, provided comments as sought by reference (1). A copy of reference (2) is enclosed herewith. We have not yet been advised of the NRC Staff's resolution of the reference (2) comments. It is our understanding that the NRC Staff plans to issue shortly a revision of the proposed policy for resumption of construction permit licensing.

Boston Edison believes that the next appropriate step is for the NRC Staff to issue the revision of the proposed policy with due consideration of industry comments provided on November 17. We would intend to review this policy including discussions with the Staff as necessary. A follow-on meeting with your Subcommittee would seem appropriate at that time. We consider a meeting with the Subcommittee on January 6 prior to completion of the above steps to be premature.

Boston Edison Company believes that the "additional measures" that would be required by the Commission's proposed plan under Option 3 with respect to siting, degraded core, and standard review plan conformance are inordinately costly, and thus inconsistent with the stated goal of the plan. We believe that these "additional measures" would result in only a marginal increase in the level of safety for Pilgrim Unit 2. We estimate that the delay engendered by the preparation, review, and adjudication of the documentation required to substantiate these "additional measures" is likely to be on the order of 1 to 1½ years with a resulting increase in the cost of Pilgrim Unit 2 in excess of $360 million. We believe that it is essential that these "additional measures" be deleted from the Commission's plan for the resumption of construction permit licensing, and that the Commission proceed on the basis of the Option 1 that was discussed in reference (1). Option 1 would impose all the pre-TMI construction permit requirements augmented by the "applicable requirements" of NUREG-0660. These "applicable requirements" are delineated in NUREG-0718. In order to make NUREG-0718 consistent with the goal of "establishing a clear statement of requirements," Boston Edison believes that it is essential that the modifications, delineated in reference (2) be made to NUREG-0718.

In reference (2) Boston Edison pointed out that degraded core is the subject of a separate, reference (3), rulemaking. We have reviewed recently funded NRC projects related to "degraded core mitigation features." The results of this review indicate that "informed decisions" cannot be made until some key research results are produced, at the earliest, two years from now. The present base of information is not sufficient to complete an individual ASLB hearing. Any "degraded core mitigation" commitment prior to the conclusion of rulemaking could be inconsistent with the results of rulemaking. The Atomic Industrial Forum Committee on Reactor Licensing and Safety has submitted, via reference (4), comments on the degraded core rulemaking as sought by reference (3). Boston Edison believes that the appropriate approach to the degraded core issue is delineated in reference (4). Delay of issuance of the Pilgrim 2 Construction Permit due to the degraded core mitigation issue is unnecessary. After the construction permit is issued and the Degraded Core Rulemaking has been completed, Boston Edison could be required to assess potential backfits of Pilgrim 2 in

accordance with 10CFR50.109. Boston Edison recommends that the requirements with respect to the degraded core rulemaking should be deleted from the Proposed Licensing Requirements for pending Construction Permit Applications and that the results of the degraded core rulemaking be applied to Pilgrim 2 in the same manner as to other licensed facilities. We believe this is responsive to the substance of your agenda for the January 6 meeting.

Since the ACRS has repeatedly recommended that the commission apply appropriate resources to timely resolution of the NTCP policy, we continue to anticipate expeditious issuance of the revised policy by NRC Staff and subsequent review with your Subcommittee, as appropriate.

Very truly yours,

J E. Howard/RWB

/cac
cc: H. R. Denton
R. A. Purple
D. A. Scaletti

BOSTON EDISON COMPANY
800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

J. EDWARD HOWARD
VICE PRESIDENT
EAP

November 17, 1980

Mr. Samuel H. Chilk, Secretary
Attention: Docketing & Service Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

RE: Notice of Proposed Rulemaking Concerning "Proposed Licensing Requirements for Pending Construction Permit and Manufacturing License Applications" (45 Fed. Reg. 65247, October 2, 1980)

Dear Mr. Secretary:

By the captioned notice, the Nuclear Regulatory Commission published and sought comment on a "notice of proposed rulemaking." This notice delineated the Commission's plan to take into account lessons learned in connection with the Three Mile Island Unit 2 (TMI) accident in the design of nuclear power plants which are the subject of pending construction permit applications.

The proposed rulemaking represents the first action by NRC to establish post-TMI licensing requirements for construction permit applications frozen since TMI and, if approved by the Commission, would authorize the resumption of NRC action on the Pilgrim Unit 2 application. The original Pilgrim Unit 2 PSAR was docketed in December 1973 and had undergone approximately six (6) years of NRC Staff review when TMI occurred. Pilgrim Unit 2 has been the subject of five (5) years of hearings; the hearing record has been closed on all issues other than TMI; and, we are awaiting a partial initial decision from the presiding Atomic Safety & Licensing Board. The design of Pilgrim Unit 2 is approximately 63% complete. Over $250,000,000 has been expended, over $100,000,000 of which is represented by equipment presently completed and in storage. In the 20 months since TMI, the delay associated with the regulatory hiatus coupled with the inflationary economic environment has resulted in an estimated increase in the cost to complete Pilgrim 2 in excess of $325 million. The current rate of cost increase due to delay is in excess of $30 million per month.

The Commission has undertaken to develop a position with respect to the set of necessary and sufficient TMI-related requirements that should be applied in the review of applications for construction permits and manufacturing licenses for nuclear power plants. In development of its current position, three options were considered:

1. Resume licensing using the pre-TMI CP requirements augmented by the applicable requirements identified in NUREG-0660.

2. Take no further action on the pending applications until the rulemaking actions described in the Action Plan have been completed.

3. Resume licensing using the pre-TMI CP requirements augmented by the applicable requirements identified in NUREG-0660 and require certain additional measures or commitments in selected areas (e.g., those that will be the subject of rulemaking).

The Commission observes that Option 1 would minimize the review and construction impact, thereby minimizing delays in reaching regulatory decisions for the planned facilities; and that Option 2 would maximize the safety improvements but would result in extensive delays and that it believes the cost of such delays are not justified provided that design flexibility can be demonstrated. The Commission elected Option 3 as "a suitable compromise between the extremes of Options 1 and 2."

Boston Edison Company believes that the "additional measures" that would be required by the Commission's proposed plan under Option 3 with respect to siting, degraded core, and standard review plan conformance are inordinately costly, and thus inconsistent with the stated goal of the plan. We believe that these "additional measures" would result in only a marginal increase in the level of safety for Pilgrim Unit 2. We estimate that the delay engendered by the preparation, review, and adjudication of the documentation required to substantiate these "additional measures" is likely to be on the order of 1 to 1½ years with a resulting increase in the cost of Pilgrim Unit 2 in excess of $360 million. We believe that it is essential that these "additional measures" be deleted from the Commission's plan for the resumption of construction permit licensing, and that the Commission proceed on the basis of the Option 1 that was discussed in the captioned notice.

Option 1 would impose all the pre-TMI construction permit requirements augmented by the "applicable requirements" of NUREG-0660. These "applicable requirements" are delineated in NUREG-0718. In order to make NUREG-0718 consistent with the goal of "establishing a clear statement of requirements," Boston Edison believes that it is essential that the following modifications be made to NUREG-0718:

THE APPLICABLE ACTION PLAN ITEMS - NUREG-0718

I.D.4 Control Room Design Standard

Since it does not appear that IEEE-566 will be amended in the near future, please change the category for the portion of this item that deals with IEEE-566 to Category 3. Delay of a pending CP while waiting for this IEEE standard to be amended is inordinately costly.

II.A.2 Siting

Pending Construction Permit applicants should not be required to make any commitment in the area covered by the Siting Rule-making. Policy expressed by Congress appears to be inconsistent with the policy contained in NUREG-0718. Congress obviously deemed it counterproductive for the Commission to apply new siting regulations to Construction Permit Applications docketed before October 1, 1979. The instructions of Congress and the NRC's implementation of those instructions by the captioned notice appear to be inconsistent. It is clear that Congress did not intend that the Commission apply these new siting regulations to Construction Permit Applications docketed before October 1, 1979; Congress intended that the Commission only apply the new siting regulations prospectively. It is requested that the category for Paragraph II.A.2 in NUREG-0718 be changed to Category IE, not applicable to plants of the type now in review.

II.B.8 "Degraded Core Rulemaking"

Pending Construction Permit applicants should not be required to make any commitment in the area covered by the Degraded Core Rulemaking. The Commission has published advanced notice of proposed rulemaking on this subject and it would be inappropriate to attempt to resolve this complex matter in an individual licensing proceeding. We believe that gross modifications in the station design that could be associated with the "molten core retention device" should be excluded from consideration. During the comment period, Boston Edison reviewed recently funded NRC projects related to these features. The results of this review indicate that "better informed decisions" cannot be made until some key research results are produced, at the earliest, two years from now. Delaying issuance of near term construction permits until these "key research results" are available would be inordinately costly and is unnecessary. After the construction permit is issued, when the Degraded Core Rulemaking has been completed, Boston Edison could be required to assess potential

backfits of Pilgrim Unit 2 in accordance with 10CFR50.109 in the same manner as other licensed facilities. Since the content of the proposed rule concerning "Interim Requirements Related to Hydrogen Control and Certain Degraded Core Considerations" (45 Fed. Reg. 65466, October 2, 1980) is already covered by other paragraphs of NUREG-0718 (I.C.1, II.B.1, II.B.2, II.B.3, II.E.4, II.F.2, III.D.1.1 & III.D.3.3), the first sentence of paragraph II.B.8 is unnecessary. Therefore, it is requested that the category for Paragraph II.B.8 in NUREG-0718 be changed to Category IE, not applicable to plants of the type now under review.

II.D.2 Research on Relief and Safety Valve Test Requirements

The two entries in Appendix D for this item should either be combined or one entry deleted.

II.F.3 Instrumentation for Monitoring Accident Conditions (RG 1.97, Rev. 2)

Please change the category for this item to Category 3. Revision 2 of RG 1.97 was not issued in August 1980; and, furthermore, it does not appear that Revision 2 of RG 1.97 will be issued in the near future. Delay of a pending CP while waiting for this Regulatory Guide to be revised is inordinately costly.

III.A.1.1 & III.A.1.2 Emergency Preparedness & Emergency Support Facilities

Two entries are in Appendix D for each of these items. They should either be combined or one entry deleted so only one set of requirements appear for each item.

We believe that Option 1, as represented by NUREG-0718 modified by the above comments, would assure that the safety lessons learned from TMI are adequately considered in the design and construction of the reactor facilities for which construction permit applications were pending prior to TMI.

The requirement that the applicant document and justify deviations from the standard review plan should be deleted from the licensing requirements for the pending CP applicants. Delay associated with this exercise would be inordinately costly, and compliance with the Standard Review Plan is not necessary, nor it is sufficient, to establish compliance with the regulations. It is settled law that an applicant need only satisfy the Commission's regulations to obtain a permit: Maine Yankee Atomic Power Co. (Maine Yankee Atomic Station), ALAB-161, 6 AEC 1003 (1973), affirmed subnom., Citizens for Safe Power V. NRC, 524 F. 2d 1291 (D.C. cir. 1975). Furthermore a staff position paper, such as a Regulatory Guide or SRP, is not a regulation and

cannot be treated as such; Gulf State's Utilities Co. (River Bend Station, Units 1 & 2), ALAB-444, 6 NRC 760, 772 (1977). In a memorandum(1) dated January 31, 1977, the Director of Nuclear Reactor Regulation specifically exempted the pending CP and ML applications from the Plan because resource expenditure could not be justified since there was no concern as to safety level established by the existing staff review. It should also be noted that, despite the fact that previous applications did not document and justify deviations from the standard review plan, the Staff has been generally successful in establishing compliance with regulations in adjudicatory proceedings. The NRC General Counsel's memorandum(2) of August 14, 1980 clearly states: "citation of a particular NRC regulation in the SRP as a support for the review requirement does not in itself show that the review requirement establishes compliance with the regulations." This memorandum further states: "the citation and cross-referencing do not show that the regulation is fully implemented. If a regulation is applicable to two different systems dealt with in two different sections of the SRP the fact that the regulation is cited in one section does not show that the regulation formed the basis for the review requirement in the SRP on the second system." The NRC General Counsel concluded that the current review procedures for determining compliance of applications with NRC regulations are legally adequate for issuance of licenses.

In conclusion Boston Edison recommends that the requirments with respect to site evaluation, degraded core rulemaking, and Standard Review Plan conformance be deleted and that the resulting plan, Option 1, be utilized to expedite the licensing process of the remaining few NTCP plants. To facilitate expediting the licensing process it is further recommended that Licensing boards be instructed that the requirements in NUREG-0718 may be litigated only to a limited extent. Specifically, boards may entertain a contention that the requirements are unnecessary, in full or in part, and they may entertain a contention that one or more of the requirements are not complied with. They may not entertain a contention asserting that requirements beyond these are necessary. The boards' authority to raise issues sua sponte should be subject to the same limitations. Contentions relating to NUREG-0718 shall be limited to those items assigned to Category 4 and Category 5, as set forth in Section III of NUREG-0718. Finally, we urge you to issue this policy expeditiously, commit appropriate Staff resources and expedite Staff review of pending construction permit and manufacturing license applications.

Very truly yours,

J Edward Howard

(1) Memo: B. C. Rusche, Director, NRR, to NRR Division Directors: "Deviations from the Standard Review Plan." Dated: January 31, 1977.

(2) Memo: L. Bickwit, General Counsel, to the Commissioners: "Compliance with Commission Regulation." Dated: August 14, 1980.

WFJH/cac

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PUBLIC SERVICE COMPANY OF OKLAHOMA
A CENTRAL AND SOUTH WEST COMPANY

P.O. BOX 201 / TULSA, OKLAHOMA 74102 / (918) 583-3611

PSO

December 30, 1980

Dr. David Okrent, Chairman
ACRS Subcommittee on Safety Philosophy,
Technology and Criteria
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Okrent:

Thank you for your invitation to the Subcommittee meeting on January 6, 1981. I plan to attend the meeting on behalf of Public Service Company of Oklahoma ("PSO"). This letter will serve as a useful means of communicating with the Subcommittee in advance of the meeting concerning NRC's proposed NTCP requirements, and we request that it be made a part of the Subcommittee's meeting record on January 6.

PSO has reviewed the questions addressed to the NTCP applicants by the Subcommittee, and we are unable to provide meaningful answers at this time. Those questions concerning the adequacy of the NRC Staff's proposals for NTCP licensing requirements cannot be answered until the Staff makes its revised proposals available to the public. As you know, the industry comments to the proposal published in the Federal Register on October 2, 1980, opposed the implementation of Option 3 in favor of Option 1. Since we are not privy to the Staff's deliberations regarding possible revision of the proposed NTCP requirements, we will not, as of January 6, know exactly what the Staff proposals are. We are hopeful, however, that the Staff will heed our comments and incorporate them in their final recommendations to the Commissioners.

The Subcommittee's questions concerning hypothetical changes in present reactor designs are also incapable of being answered at the present time. PSO has not initiated any plant-specific engineering or research effort concerning such problems as quantifying a safety goal or revising reactor design parameters to respond to various hypothetical degraded core conditions. PSO is, however, participating in the broad industry effort to be spearheaded by the Industry Degraded Core Rulemaking ("IDCOR") group. IDCOR is dedicated to participating in the rulemaking and assisting the NRC in developing a safety goal and in re-examining reactor design

bases with respect to degraded core conditions. These matters are complex and reasoned engineering judgments and conclusions will only be realized after significant technical analyses are performed by the NRC and the industry. In this connection, we understand that IDCOR is contemplating a substantial research effort to help find answers to questions such as those asked by the Subcommittee.

It might be helpful to the Subcommittee if I summarize PSO's position with respect to the NRC Staff's NTCP proposal of October 2, 1980. Our detailed comments are attached for your information. However at the risk of over-simplification I would characterize our basic objection to Option 3 as one of timing, i.e., an option that requires the completion of a myriad of time-consuming engineering activities and analyses before issuance of construction permits. On the other hand, Option 1, which we believe imposes essentially the same technical requirements as Option 3, would require only that an applicant make necessary commitments, including reasonable implementation schedules, before issuance of construction permits. Consistent with the regulatory treatment being accorded existing CP holders by the NRC, the engineering details of implementation would be completed during the post-construction permit period but prior to the issuance of any operating license rather than prior to construction permit issuance as contemplated under Option 3. The Option 1 approach makes emminent good sense when one recognizes that meaningful answers to the degraded core and safety goal questions involve fundamental design questions that cannot be answered adequately without thoughtful consideration over the next 2 to 4 years. None of the NTCP projects will remain viable if construction permits are held up until then. Any notion that these questions can be answered in the near future -- and it is possible some members of the NRC Staff think so -- will only result in contradictory answers and interminable delays before NRC licensing boards.

For these reasons, we urge the Subcommittee to recommend the adoption of Option 1 with the clear understanding that the technical issues which concern the Subcommittee will be adequately addressed during the post-construction permit period but prior to the issuance of any operating license.

Sincerely,

Vaughn L. Conrad

Vaughn L. Conrad
Manager, Licensing & Compliance

Enclosure

PUBLIC SERVICE COMPANY OF OKLAHOMA
A CENTRAL AND SOUTH WEST COMPANY

P.O. BOX 201 / TULSA, OKLAHOMA 74102 / (918) 583-3611

PSO

November 17, 1980

Samuel J. Chilk
Secretary to the Commission
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Docketing and Service Branch

Public Service Company of Oklahoma ("PSO") submits the following comments in response to the NRC Notice of Proposed Rulemaking* published in the Federal Register on October 2, 1980, (hereinafter referred to as the "Notice"). This Notice, which is entitled "Proposed Licensing Requirements for Pending Construction Permit and Manufacturing License Applications", would directly affect PSO's application (on behalf of Associated Electric Cooperative, Inc., Western Farmers Electric Cooperative and itself) for permits to construct and operate the Black Fox Station (U.S. NRC Docket No. STN-556 and 557) which consists of two 1150 Mwe boiling water reactors to be located near Tulsa, Oklahoma.

Regulatory action on PSO's application which has been pending before the NRC since the winter of 1975, has been suspended since March 28, 1979 -- the date of the Three Mile Island Unit 2 (TMI-2) accident. Since that date, no progress has been made by the Atomic Safety and Licensing Board toward issuing its decision on the pre-TMI hearing record (which was closed February 28, 1979) or by the NRC staff toward reviewing PSO's commitment to implement the lessons learned from TMI-2 into the construction and operation of the Black Fox Station. The proposed rulemaking represents the first action by NRC to establish post-TMI licensing requirements for construction permit and manufacturing license applications frozen since TMI and, if approved by the Commission, would authorize the resumption of NRC action on the Blac Fox application.

*45 Fed. Reg. 65247

The Commission has considered three options with respect to the proposed rulemaking initiative, two of which it rejected. Option 1, one of the rejected options, would have permitted the resumption of licensing using pre-TMI CP requirements as supplemented by a commitment to implement, during the post-CP period, the applicable task action plan items described in the NUREG-0660, "NRC Action Plans developed as a result of the TMI-2 accident." As noted in NUREG-0718, "Proposed Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," Option 1 would have "treat[ed] pending CP and ML applications as if they were the last of the present generation of nuclear plants". Option 1 has, in effect, been applied to existing CP holders since facility construction under these licenses has not been suspended and the requirements of NUREG-0660 are being implemented during course of the post-CP and OL licensing review for these projects.* Although this option clearly presented the most promise for minimizing undue delays in reaching regulatory decisions for applications like Black Fox without compromising implementation of the task action plan items, the Commission rejected Option 1 because of what it considered to be the disadvantage of possibly not maintaining available design flexibility for such applications.

NRC also rejected Option 2, a formal moratorium on renewed licensing, until long-term rulemakings are completed because this option involved unwarranted delays in the consideration of pending CP and ML applications. NUREG-0718 notes that the selection of Option 2 would have "treat[ed] the pending CP and ML applications as the first of a new generation of nuclear plants".

Instead, the Commission endorsed Option 3 because, in its view, this Option would present "a suitable compromise between the extremes of Option 1 and Option 2". Option 3 would permit the resumption of licensing but only after satisfying, in addition to the pre-TMI CP requirements, the implementation, in whole or in part, of many of the task action plan measures prior to the issuance of construction permits.

PSO concurs in the Commission's rejection of Option 2. Since the degraded core rulemaking is likely to take two to four years to complete, the impact of the continued moratorium implicit in Option 2 is unacceptable both to the national goal of energy independence, and to the economic viability of projects like Black Fox Station. Since PSO, as the holder of a Limited Work Authorization, has expended in excess of $250,000,000

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*This licensing plan is set forth in NUREG-0660 as augmented by NUREG-0694 "TMI-Related Requirements For New Operating Licenses," and the NRC's Policy Statement published in the Federal Register on June 20, 1980.

in engineering and equipment, site preparation, excavation and other sunk costs, the carrying charges for this investment along with the rising cost of inflation represents a high and continuous cost burden on the Black Fox project. The unknown length of the delay and escalation of design requirements inherent in Option 2 would create serious questions as to the viability of the Black Fox project, thereby requiring PSO to consider alternatives, such as conversion of the project to a coal-fired facility in order to meet the system load requirements in the late 1980's.

Unfortunately Option 3 would not relieve the increasing cost pressures on the Black Fox Project. Option 3 like Option 2 would extend and protract the licensing process for applications like Black Fox. Site re-analyses, degraded core analyses and redesign work and the additional engineering effort needed to satisfy the remaining task action plan items prior to the issuance of construction permits, all promise to extend the review, evaluation and hearing time needed for the consideration of applications like Black Fox beyond any measure of reasonableness. Option 3 does not clearly and adequately reflect a definitive licensing path which bounds requirements appropriate to a construction permit proceeding. Nor is it, as the notice states, "relatively easy to provide design flexibility to implement potential significant safety improvements" when one is working in the real world of facility design and construction so the proposed rule fails in its quest to introduce stability into the CP review process. PSO believes that Option 3 in its present form contains corporate risks identical to Option 2 regarding the uncertainty of station design requirements.

Therefore, the result on station construction schedule is equivalent to Option 2, since in order to properly limit those risks the delay would be self-imposed by the Company rather than mandated by the NRC.

Hence, PSO must respectfully take issue with the Commission's assessment since we firmly believe that, with the present composition of the various options set forth by NRC, Option 1 represents the proper approach to proceeding with the licensing of Black Fox Station if that project is to remain within acceptable corporate risk boundaries. This option is completely consistent with the Agency's treatment of the projects presently holding construction permits. Those projects have been allowed to continue construction in the wake of the events of Three Mile Island. The Black Fox design, by the very nature of being more current than the designs of plants under construction, has incorporated certain improvements not included in earlier units now under construction. Furthermore, PSO has, just as if it were a permit-holder, voluntarily committed* on

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*PSO submitted project specific analyses and commitments by correspondence dated June 15, July 27, and August 24, 1979 addressing the TMI-2 events as they related to Black Fox Station.

the docket to implement the "lessons learned" from TMI-2. Indeed, in August 1979 PSO called for a re-opening of the Black Fox licensing hearings to publicly demonstrate this commitment.

The only reason stated for NRC's rejection of Option 1 was a perceived need to assure design flexibility to accommodate the content of the rule that might result from the long-term degraded core rulemaking. However, the NRC didn't deem this requirement -- to preserve design flexibility -- necessary for plants under construction. If foreclosure of possible options resulting from the degraded core rulemaking were a real safety concern, NRC would have issued orders suspending construction under existing CP's and directed that such permit holders perform the design exercise for degraded core matters described under Option 3. This not being the case, it can only be concluded that the imposition of these NUREG-0660 measures as a pre-licensing requirement for pending CP and ML applications is arbitrary and discriminatory.

Like those units under construction, Black Fox should properly be treated as the last of the present generation of nuclear power plants. The proposed Black Fox Station consists of two BWR-6, Mark III machines. The design for this facility has been reviewed and evaluated under NRC's Standard Review Plan concept and the nuclear steam supply system has been reviewed as a part of NRC's standardization program. A preliminary design approval for the BWR-6 NSSS has been granted the General Electric Company on the GESSAR-238 docket. Thus it is apparent that Black Fox is of a vintage identical to the Grand Gulf and the Hartsville nuclear projects and other facilities presently under construction. The Black Fox application would receive, under Option 1, regulatory treatment consistent with that being implemented by NRC for these facilities. The approach proposed under Option 3 would, without justification, deprive PSO of treatment equal to holders of construction permits similarly situated except for a quirk of timing and it should therefore be rejected.

The discrimination inherent in such treatment becomes more evident when one considers the paucity of Staff resources being devoted to CP licensing. By focusing virtually all the Agency's resources away from pending construction permit applications, the NRC has saddled the Near-Term Construction Permit companies with an additional delay which could not have been reasonably foreseen by even the most omniscent corporate executive. This delay has had the discriminatory effect of propelling the Black Fox Station into the "new wave" of regulatory requirements, which will be applied to the next generation of facilities, while holding our application in limbo. The denial of PSO's equal right to the processing of its application has thereby introduced serious new questions concerning project viability.

It is clearly in the public interest to resume CP licensing, and we applaud the NRC for its action, albeit long overdue. We recognize that licensing resumption must be on a basis that assures that the lessons learned and to be learned from the accident at TMI are considered in the design and construction of reactor facilities. In the case of Black Fox Station, we are convinced that Option 1 can clearly satisfy this objective. Therefore, we urge the NRC to reopen its decision as to the proper course of action for the Black Fox Station licensing docket.

Although it is PSO's position that Option 1 provides the only proper plan for licensing Black Fox Station, we feel it is incumbent on us to comment in detail on Option 3 since the Agency has publicly adopted this plan; further to demonstrate the risks added to a project by Option 3 and the Agency's total lack of sensitivity to the unique position of the NTCP group, especially Black Fox Station, due to its completeness of design and licensing status.

1. Siting

This special provision states that CP applicants must (i) compare their sites with the recommendations of NUREG-0625, as amended by NRC's Office of Policy Evaluation and the ACRS, and (ii) assess their applications against the criteria of the proposed rule on siting (scheduled for issuance in October 1980). This statement of siting policy, which is intended by the terms of the Notice to apply to CP applications filed before October 1, 1979, is inconsistent with law -- Section 108 of the NRC Authorization Act for fiscal year 1980 -- and NRC's "Advance Notice of Rulemaking: Revision of Reactor Siting Criteria", (45 Fed. Reg. 50350, dated July 29, 1980). Both the Authorization Act and the Advanced Notice on Siting clearly declare that the new siting policy and criteria now under development are not to be made applicable to CP applications filed prior to October 1, 1979. Moreover, contrary to the statement in the Notice that the proposed new rule is scheduled for issuance in October 1980, PSO now understands that the Office of Standards Development does not intend to issue such a rule until some indefinite but extended time in the future.

Although the application of new siting rules and criteria to applications like Black Fox is impermissible, under the law, the NRC may nonetheless, as a matter of discretion, review the sites for such applications in the manner contemplated by the Advance Notice on Siting (45 Fed. Reg. 50350-51). Therefore, PSO suggests that the section in the Notice on Reactor Siting be rewritten along the following lines:

Siting

The NRC Staff will review existing sites for pending construction permit applications in order to examine whether additional modifications in operating procedures, design or equipment might be necessary. Such a discussion will be included in the Staff's Safety Evaluation Report (SER) or in a supplement to the SER.

2. Degraded Core Rulemaking

It appears from this special provision that NRC is incorporating, as a part of its pre-CP review, requirements for obtaining appropriate compliance with the interim rule on degraded core considerations (45 Fed. Reg. 65466 et. reg. October 2, 1980) and for appropriate accommodation of the results of the rulemaking on the final rule. With respect to the latter requirement, pending CP applications could be reviewed to assure that to the extent practicable and feasible, existing reactor designs take full advantage of available design flexibility to avoid foreclosure of specified degraded core prevention or mitigation measures that could result from or provide a risk reduction consistent with the final rule. NRC cannot intend by this special provision to mean that feasibility encompasses requiring applicants to discard completed engineering and design work (more than 50% of the design on the Black Fox Station has been completed). Rather it must intend that steps be taken within the parameters of existing design concepts to avoid foreclosure of specific degraded core measures by construction of the reactor facility. For example, if space for a particular measure does not exist, and if it could only be accommodated by a radical redesign of the plant, then such a measure would not be deemed to be "practicable" for design modification purposes and could be discarded as a potential design feature of the plant. Likewise, if a particular measure is technically beyond the commercial state of the art, it would not be considered practicable or feasible. On the other hand, it may be feasible to avoid foreclosure by expending reasonable time, effort and monies to make changes within the context of presently existing designs. This special provision on "degraded core rulemaking" should only provide NRC with the mechanism to scrutinize and regulate design decisions that are effectively being made today on these matters.

The special provision as it appears in the notice is at best ambiguous and does not clearly reflect the foregoing. That text should be changed to read as follows:

Degraded Core Rulemaking

CP and ML applications would comply with the applicable requirements of the interim rule.

CP and ML applicants would also be required to perform an analysis which would review and evaluate their reactor designs for the purpose of identifying the optimal additional practical measures, both preventive and mitigative, that have the potential for significant risk reduction. Applicants would commit not to foreclose these measures by construction of the plant. The NRC Staff would review these analyses to determine, with reasonable assurance, that taking into account appropriate facility design considerations and the relevant state of the technology facility construction would not foreclose the installation of those potential modifications to the facility.

These potential requirements may include such features as filtered vented containment, molten core retention, and hydrogen control systems. Installation of such features would be done only if the resulting risk reduction were required as a result of the final rulemaking.

3. Reliability Engineering

The utilization of reliability analyses for nuclear power plant evaluation is an especially active analytical area at the present time, with many projects under way. In such a dynamic era the methodology is likely to change significantly. This is especially true in the treatment of operator error and common cause contributions to system and plant reliability. It is likely that any planned reliability programs proposed by construction permit applicants would undergo marked changes before implementation due to the evolutionary state-of-theart and the present lack of a safety goal definition. Thus, the requirement to define a reliability program in other than the most general terms would be unproductive at the construction permit application stage.

The quantitative results of reliability analyses are highly dependent on plant specific designs and individual vendor supplied equipment. Although BFS design is over 50% complete at the CP stage vendor specific equipment information and/or operating procedures are generally not available. Thus, reliability analyses performed early will contribute little more than those analyses which will be available from current plants either operating or in the final stages of construction. Design decisions based on preliminary design data may in fact suggest design changes that are unnecessary or undesirable when evaluated against final design information.

Successful utilization of reliability studies depends upon both the integrity of the analyses and upon the existence and validity of reliability goals and derived criteria for specific systems and equipment. The NRC has only recently published a plan for developing a safety goal from which equipment and systems reliability criteria may be determined. A commitment to factor reliability studies into designs can only be justified when the methodology is reasonably mature and consistent, when safety and reliability goals are established and when the cost-benefit results can be used to identify reasonable solutions to problem areas. No prudent manager could commit to design changes or plant "back-fits" until clear goals and consistent criteria are defined.

PSO believes that reliability analyses can be important tools to help assure not only the safe design of BFS, but also the safe and reliable operation of the facility. PSO is convinced that it is premature to require detailed definitions of a program, including methodology and schedule, because safety goals are as yet undefined in conjunction with the developmental statue of reliability engineering. There is a significant time remaining between CP issuance and conclusion of design would would permit development and implementation of a reliability program in a prudent engineering manner.

4. Emergency Preparedness

The Notice stated that construction permit applicants would submit, prior to the issuance of construction permits, a discussion of their preliminary plans for coping with emergencies. The purpose of the report would be to address the revised rules and regulations published in the Federal Register August 19, 1980, revising Sections 50.33, 50.54, and Appendix E, along with addressing a new Section, 50.45 to Title 10 of the Code of Federal Regulations. These rules became effective on November 3, 1980. In particular, Section II of Appendix E has been revised requesting specific information on the new emergency response planning bases for the preliminary safety analysis report stage.

PSO notes that the information requested on this section of the proposed policy specifically addresses the PSAR portion of the new rules and is not intended to represent information to be supplied in an operating license emergency response plan.

5. Action Plan

The Notice stated that action plan items determined by the NRC to be applicable to CP and ML applications are set forth in NUREG-0718. It appears that this letter may be PSO's only opportunity to comment on NUREG-0718. With the exception of the matters discussed below, PSO believes the requirements of that document are clearly stated.

PSO believes that NRC should allow maximum advantage to be taken of the long term activity of the nuclear industry in various groups within the industry. This appears to be relatively easy for the Category 2 items which only require a commitment to implementation prior to OL issuance; however, for Categories 3, 4 and 5 items, it appears it will be more difficult to meet the requirements of NUREG-0718 and at the same time take advantage of work being done by the various industry groups such as AIF, INPO and NSAC, as well as the various owners' groups. PSO is participating in several of these in order to develop the best answers to the lessons learned from TMI. Some specific examples of these conflicts are discussed in Appendix A. Additionally, NRC must provide the necessary evaluation criteria for the area of management for design and construction.

6. Standard Review Plan Compliance

In addition to the foregoing special provisions, the Notice mandates that CP and ML applicants identify and justify the bases for all deviations from the acceptance criteria set forth in the May 1980 revision of the NRC Standard Review Plan ("SRP") and NUREG-0718 prior to issuance of any construction permit or manufacturing license. This requirement would apply to all CP and ML applications for which NRC Staff SERs or SER TMI supplements are issued before January 1, 1982. Applications for which SERs etc. were issued after January 1, 1982 would be compared to the next revision of the SRP (currently expected to be issued in April 1981).

PSO strenuously objects to the application of this new requirement prior to the decision on whether or not to issue construction permits for the Black Fox Station because no enhancement of reactor safety is achieved. Instead, this pre-CP requirement delays even further an NRC determination on the Black Fox application. Such undue delays further submit PSO and its ratepayers to the ravaging effects of inflation and interest on funds expended.

The delay results from two causes. First, the exercise of comparing the SRP with the Black Fox PSAR will add at least eight months to the review process in terms of the time needed by PSO and the NRC Staff to complete their reviews. Since NRC, by its own plan has only limited resources to deal with CP applications (NRC does not even have a project manager assigned to Black Fox at this time), it is impossible to assess the real delays which would be encountered in this extensive review. Second, the process of identifying "deviations" from the SRP provides a ready invitation to intervenors and NRC Boards to reopen safety issues that have been and remain closed in the Black Fox proceeding and in others like it. Such delays including the revisiting of closed issues might be warranted if a commensurate safety benefit were realized; but as explained below,

the SRP review exercise will provide very little, if any, improvement in reactor safety for the Black Fox Station.

PSO estimates it will take a minimum of three months of engineering time to simply identify deviations from the SRP. Thereafter it will take an additional three to five months of engineering effort to review the Black Fox design -- now more than 50% complete -- to document the significance of and justification for such deviations and to prepare a PSAR amendment for submission to the Staff. PSO cannot estimate how long the NRC Staff will require to review the PSAR amendment given the present attitude of resource allocation. If adequate resources were made available in a timely manner, the NRC would need a minimum of three months review and SER supplement preparation time.

Once the PSAR amendment is submitted to the NRC, the Staff will be forced to revisit areas of its technical evaluation that have been completed for the Black Fox application simply to ascertain formal conformity or nonconformity with the SRP. The NRC Staff would necessarily re-examine licensing decisions made in 1975 through 1979 concerning the efficacy of the Black Fox design. Undoubtedly the NRC would reaffirm that those decisions remain sound today simply because not much has changed other than the additional TMI-related requirements.

The NRC Staff's safety review on the Black Fox docket has been exhaustive. This review has spanned four years, and during the process the Staff has asked several hundred questions, answers to which have been documented in the PSAR. The guidance set forth in regulatory guides in both draft and final form and branch technical positions has been factored into the Black Fox docket. The Staff safety evaluation in the context of its TMI review is still ongoing and indeed the Black Fox application remains subject to the incorporation of any significant new safety requirement. This ongoing effort -- commenced in 1975 -- provides ample evidence that the NRC Staff has conducted and is conducting its safety evaluation in a responsible and effective manner. The redundant SRP review would add nothing to this impressive record.

As a pre-licensing requirement, the SRP review under NRC's Rules of Practice affords an open invitation to intervenors to inquire into each and every matter, and in any event imposes an obligation on NRC Boards to determine the adequacy of the Staff's safety judgments regardless of the depth and sufficiency of their review. This resulting delay is both inequitable and unwarranted with respect to the Black Fox docket, for this is not a virgin proceeding. Six weeks of safety hearings have been held and completed. The hearing record is closed. Although PSO expects to reopen the hearing record to consider TMI-related

matters, the SRP review as a pre-CP requirement permits a total re-opening of the entire safety review -- a result that has the effect of essentially starting over on the BFS licensing process. This should not be authorized by the Commission without the strongest justification.

The previous Staff review of the Black Fox Station was conducted against the 1976 version of the SRP. This prior review was similar to that proposed in the Notice except that the Staff did not kept a neat scorecard of its review so that it could easily certify compliance with the SRP. If the Commission desires to change the NRC's current techniques for the safety evaluation of nuclear power facilities, as suggested in the general SRP rulemaking notice on October 9*, this new approach should be introduced for pending CP applications as a post-licensing consideration in the same manner proposed by the Commission for pending operating license applications and it should be done against the updated revised SRP. Another review for Black Fox against the same SRP utilized in 1976 seems pointless. As the general SRP notice indicates, pending operating license applications where it is expected that SERs or TMI SER supplements will issue before January 1, 1982 would be required to conduct the SRP review after the issuance of the operating license. Surely if valid doubt exists concerning the adequacy of the Staff's safety reviews, the Commission would have proposed the SRP review as a pre-license condition for pending operating license applications as well as for similarly situated CP applications. We believe the course of action proposed for pending OL applications is consistent with the public health and safety and interest. No sound reason exists to maintain a contrary policy for pending CP applications and to do so is arbitrary and capricious.

It should be emphasized that the Commission's actions concerning the SRP review of pending CP applications involve solely matters of policy. No statute or regulation requires this SRP review exercise. Indeed, Section 110 of the NRC FY80 Authorization Act -- the genesis of the SRP requirement -- applies only to operating reactors, and since that fiscal year has expired and if it is true that NRC is no longer using FY80 monies, PSO doubts that Section 110 imposes any continuing legal obligation on NRC to continue the exercise of SRP/FSAR comparisons. Nevertheless, PSO recognizes that the Commission may establish, as a matter of policy, different methods to facilitate its Staff's safety reviews. However, in effecting this change the Commission must exercise its discretion in a reasoned manner. The proposal to require the SRP review as a pre-license requirement for CP and ML applications does not

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*See Notice of Proposed Rulemaking: Plan to Require Licensees and applicants to Document Deviations From the Standard Review Plan, 45 Fed. Reg. 67099, dated October 9, 1980.

meet this test. The proposal should be changed to a post-licensing consideration.

Finally, views are requested with respect to the extent to which judgments reached by the Commission on requirements for CP applicants should form the basis for instructions to NRC licensing and appeal boards in CP and ML proceedings. As the Commission recognized for pending operating license applications*, NRC Boards will also require Commission direction with respect to the resumption of CP and ML licensing proceedings.

PSO believes that the Commission should issue a rule in conformity with the terms of Option 1. NRC Boards should be directed to resume licensing proceedings in accordance with this rule. We note, however, that such a rule would depart substantially from the thrust of the Notice as now constituted. As a result, further notice and opportunity for comments appears to be necessary to satisfy the rulemaking requirements of Section 553 of the Administrative Procedures Act by giving reasonable notice to members of the affected public of the actual contents of the plan to resume CP and ML licensing activities. If, on the other hand, the Commission decides to implement Option 3 either as it stands or as modified by the foregoing specific comments, PSO believes that the notice to the public already given is legally sufficient to support the immediate issuance of a rule. Consequently, the plan should be embodied in a rule effective within 30 days of publication in the Federal Register.

In summary, PSO's position, giving due weight to both the critical need for the Black Fox Station project on Applicants' electrical systems and the financial commitment to date as balanced by the evolving spectre of undue corporate risk is that Option 1 represents the only proper approach to proceeding with the project as it is presently constituted. We should, by right, be treated consistently with existing construction permit holders. Adoption of any other Option by the Commission clearly signals that we are being treated as the first of a new generation of nuclear fueled electric generating facilities. This inequity creates a corporate risk beyond any envisioned when the Black Fox application was tendered.

In the event that the Commission rejects our request for Option 1 and instead elects Option 3, we will carefully evaluate our course of action based upon the "final" Option 3 adopted and published by the Agency.

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*See Further Commission Guidance for Power Reactor Operating Licenses: Statement of Policy, 45 Fed. Reg. 41738, dated June 20, 1980.

PSO appreciates this opportunity to provide these written comments. We would request the opportunity to orally address the Commission during its further consideration of the Notice.

Sincerely,

Vaughn D. Conrad
Manager, Licensing & Compliance

Appendix A

I.B.1.1. - Organization and Management Long-Term Improvements

This item is an activity currently underway by INPO with the expected completion to be by 5/81. On this schedule it will be impossible to take advantage of the work being done by INPO and at the same time submit a timely response to this item. Any response by the NTCP utilities may not be compatible with the industry position and thereby prejudice the Staff prior to review of the industry response or require an additional submittal from the applicants to obtain compatibility with the industry position. In any case, the requirements of item I.B.1.1 should be focused more toward the early construction management activities with a required commitment to review the industry position and develop a program implementing those activities which are required to control the construction project in the latter stages and achieve a smooth transition into the operation phase.

I.C.9 - Long-Term Program Plan for Upgrading of Procedures

NUREG-0718 calls for the scope of the program to include emergency procedures, reliability analysis, human factors engineering, crisis management and operator training. The requirement goes on to state the applicants shall also describe how their program will be coordinated with INPO activities. This item is an activity currently underway by INPO with a projected completion of 1/82. This schedule will make it difficult for applicants to determine in any significant detail to what extent their program will interface with the program being developed by the industry.

I.E.4 - Coordination of Licensee, Industry and Regulatory Programs

The NUREG-0718 requirements indicate that the applicants shall, in conjunction with Action Plan Item I.C.5, provide a description of their program to evaluate experience both at their own plant and similar plants and factor this experience as appropriate into the design and construction of their plant. In addition, the program shall describe how these activities will be factored into the operation of the plant. This item is curently an activity underway by NSAC and no projected schedule is available at this time. This particular item is a perfect example of an activity in which the construction permit applicants are being required to submit the details of the

program which is a generic industry item. This requirement not only imposes additional and unnecessary work on the utilities, it also holds the potential of an individual utility submitting a response which may undermine the program being developed by the industry.

II.F.3 - Instrumentation for Monitoring Accident Conditions (Regulatory Guide 1.97)

This item requires the applicant to provide in their facility design instrumentation to monitor plant variables and systems during and following an accident in accordance with design bases that are defined and criteria requirements that are specified in Regulatory Guide 1.97. NUREG-0718 also indicates that Regulatory Guide 1.97 should be issued in final form by August 1980. However, this Reg. Guide has not been issued and is currently being reviewed in Draft 3 to Revision 2 of this Regulatory Guide. Because of this delay in issuing Regulatory Guide 1.97, it will be difficult for the utilities to meet the requirements as specified in NUREG-0718 Item II.F.3 in a timely manner.

II.J.3 - Management for Design and Construction

This item is closely allied with Item I.B.1.1. The objective of this item is to improve the qualification of licensees for operating nuclear power plants by requiring greater oversight of design construction and modification activities. NRR is to develop criteria requiring license applicants and licensees to improve their oversight of design and construction. The Staff indicates by the schedule in NUREG-0660 that they will issue a Regulatory Guide for comments by March 1981 with the final guide by October 1981. PSO believes this area is being ignored for construction permit applications, as evidenced by the recently released NUREG-0731 entitled, "Guidelines for Utility Management Structure and Technical Resources", which made no mention of management for design and construction, as opposed to earlier drafts of the document which had included some very general criteria.

We feel that it would be in the best interest of both the industry and the NRC to take maximum advantage of the other industry groups' efforts and delete the duplication of efforts as outlined above for items I.B.1.1, I.C.9, II.F.1 and II.J.3. By so doing we can conserve everyone's strained resources and also allow the industry to provide an unbiased work product on a schedule developed with appropriate durations and priorities.

NUREG-0718

JAN 0 5 1981

DRAFT

Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License

Final Report

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

LICENSING REQUIREMENTS FOR PENDING APPLICATIONS FOR CONSTRUCTION PERMITS AND MANUFACTURING LICENSE

NUREG-0718

DRAFT

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Final Report

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

The TMI-2 Action Plan, NUREG-0660, does not specifically address requirements for construction permit and manufacturing license applications. There are currently pending six construction permit applications for eleven plants and one manufacturing license application for eight floating nuclear plants. Staff review of these applications has been suspended since the TMI-2 accident pending the formulation of a policy to appropriately reflect the lessons learned from the accident. This report summarizes the TMI-related requirements that are to be applied to these seven applications.

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I. INTRODUCTION

After the accident at Three Mile Island, Unit 2, on March 28, 1979, the Commission directed its technical review resources to assuring the safety of operating power reactors rather than to the issuance of new licenses. Furthermore, the Commission decided that power reactor licensing should not continue until the assessment of that accident had been substantially completed and comprehensive improvements in both the operation and regulation of nuclear power plants had been set in motion.

Following the accident at Three Mile Island, Unit 2, the President established a Commission to make recommendations regarding changes necessary to improve nuclear safety. In May 1979, the Nuclear Regulatory Commission established a Lessons Learned Task Force to determine what actions were required for new operating licenses and chartered a Special Inquiry Group to examine all facets of the accident and its causes. These groups have published their reports.

The Lessons Learned Task Force led to NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report." Following release of the report of the Presidential Commission, the Commission provided a preliminary set of responses to the recommendations in that report. This response provided broad policy directions for development of an NRC Action Plan, work on which was begun in November 1979. During the development of the Action Plan, the Special Inquiry Group Report was received, which had the benefit of review by panels of outside consultants representing a cross section of technical and public views. This report provided additional recommendations.

NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," was developed to provide a comprehensive and integrated plan for the actions judged appropriate by the Nuclear Regulatory Commission to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at Three Mile Island, Unit 2, and the official studies and investigations of the accident. In developing the Action Plan, the various recommendations and possible actions of all the principal investigations were assessed and either rejected, adopted or modified.

Actions to improve the safety of nuclear power plants now operating were judged to be necessary immediately after the accident and could not be delayed until the Action Plan was developed, although they were subsequently included in the Action Plan. Such actions came from the Bulletins and Orders issued immediately after the the accident, the first report of the Lessons Learned Task Force issued in July 1979, the recommendations of the Emergency Preparedness Task Force, and the NRC staff and Commission. Before these immediate actions were applied to operating plants, they were approved by the Commission. Many of the required immediate actions have already been taken by licensees and most are scheduled to be complete by the end of 1980.

On February 7, 1980, based on its review of initial drafts of the Action Plan, the Commission approved a listing of near-term operating license (NTOL) requirements, as being necessary but not necessarily sufficient TMI-related requirements, for granting new operating licenses. Since then, the fuel load requirements on the NTOL list have been used by the Commission in granting operating licenses, with limited authorizations for fuel loading and low power testing, for three plants.

On May 15, 1980, after review of the last version of the Action Plan, the Commission approved a list of "Requirements for New Operating Licenses," now contained in NUREG-0694, which the staff recommended for imposition on current operating license applications. That list was recast from the previous NTOL list and sets forth the TMI-related requirements and actions for new operating licenses. In a Statement of Policy issued on June 16, 1980, the Commission concluded that the list of TMI-related requirements for new operating licenses found in NUREG-0694 is necessary and sufficient for responding to the TMI-2 accident. The Commission has decided that current operating license applications should be measured against the regulations, as augmented by these requirements. Subsequently, the staff incorporated all of the TMI-related items for operating reactor licensees and operating license applicants in one document, NUREG-0737, which was reviewed and approved by the Commission on October 29, 1980. This report was issued by letter on October 31, 1980.

The TMI-2 Action Plan, NUREG-0660 does not specifically address requirements for construction permits (CP) or manufacturing license (ML) applications. There are currently pending six CP applications for eleven plants and one ML application for eight floating nuclear plants. The NRC staff review of these applications has been suspended since the TMI-2 accident pending the formulation of a licensing policy to appropriately reflect the lessons learned from the accident. Therefore, the NRC staff initiated a program to propose for Commission approval a course of action that would lead to the establishment of TMI-2 related requirements for these applications. This report is the result of that program and the sections that follow describe the TMI-related requirements for the pending construction permit and manufacturing license applications.

II. ASSESSMENT OF TMI-2 ACTION PLAN FOR PENDING CP AND ML APPLICATIONS

In order to assess the extent to which the TMI-2 Action Plan should be implemented on the seven pending CP and ML applications, the staff developed five requirement categories. Each of the TMI-2 Action Plan requirements were carefully evaluated and then assigned to one of these five categories. A discussion of each of the requirement categories follows.

Category 1

A requirement of a type not applicable to the pending CP or ML applications for any of the following reasons:

a. It can only be addressed in operating license applications or by licensees;

b. It is not directed to CP or ML applicants;

c. It does not apply to plants of the type now pending;

d. It has been (or will be) superseded by a more restrictive requirement in the Action Plan;

e. It has already been completed.

Category 2

A requirement of the type customarily left for the operating license stage. The applicant should indicate its recognition of the need for development of operating license or final design requirements and should provide a commitment to implement such requirements in connection with its application for approval of the final design.

Category 3

Studies (and other research and development activities to provide design development information) of the type customarily left for review at the final stage. However, to satisfy 50.35(a)(3) the staff believes that items in this category should be completed as early as is practicable so that the results can be most effectively taken into account in developing final design details. The applicant should provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and a program to assure that the results of such studies are factored into the final design.

Category 4

A requirement to demonstrate that any additional design, development and implementation necessary to satisfy the requirement (or to satisfy the goals of the task whose requirements are to be developed in the future) will be satisfactorily completed by the operating license stage. This is the type of information customarily required at the construction permit stage to satisfy 50.35(a)(2), or to satisfy ALAB-444 with respect to generic issues.

Category 5

A requirement for information of the type customarily reviewed at the preliminary design stage for the following types of items:

a. Items for which the required information should be sufficient to demonstrate that the requirement has been satisfied by the application. This is the kind of information and degree of detail customarily provided at the preliminary design stage with respect to site and major systems and structures to satisfy 50.34(a)(1). This will also be applicable to items relating to technical qualifications of the applicant and its management for design and construction.

b. Items for which the required information should be sufficient to assure that the requirement will be met at the final design stage. This is the kind of information and degree of detail customarily provided at the preliminary design stage with respect to the preliminary design of the facility to satisfy 50.34(a)(3)(4), etc.

It should be noted that in assigning individual TMI-2 Action Plan requirements to one of the aforementioned categories, the NRC staff did not limit itself to a narrow interpretation of the Action Plan requirements. Rather, we took into account the fact that much work could be done by the applicants to address the specific concerns of the individual Action Plans. In such cases we defined the specific concerns that should be addressed by the CP and ML applicants and the level of information to be supplied in order for the staff to verify that the requirement has been (or will be) satisfied.

Tables 1, C.1, C.2, and C.3 from NUREG-0660 list each of the TMI-2 Action Plan requirements. Appendix A of this report is a reprint of these tables with the NRC staff's category assignments for the pending CP and ML applications.

Appendix B provides a cross-cut of the Action Plan requirements by category assignment for the pending CP and ML applications.

Appendix C provides a description of the specific information to be provided by CP and ML applicants for each of the Action Plan requirements assigned to Category 3, 4, and 5.

APPENDIX A

REQUIREMENT CATEGORY ASSIGNMENTS

FOR PENDING CONSTRUCTION PERMIT AND MANUFACTURING LICENSE APPLICATIONS

TABLE 1 - PRIORITIES AND STATUS OF ITEMS IN TMI-2 ACTION PLAN

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
I.	Operational Safety		
I.A	Operating Personnel		
I.A.1	Operating Personnel and Staffing		
	1. Shift Technical Advisor	2/1c	
	2. Shift Supervisor Admin. Duties	2/1c	
	3. Shift Manning	2/1/c	
	4. Long-Term Upgrading	1b/1b	Refer to Action Plan Item I.B.1.1
I.A.2	Training and Qualifications of Operating Personnel		
	1. Immediate Upgrading of Operating and Senior Operator Training and Qualifications	1d/1c	Refer to Action Plan Item I.B.1.1
	2. Training and Qualifications of Operations Personnel	1d/1c	Refer to Action Plan Item I.B.1.1
	3. Administration of Training Programs for Licensed Operators	1b/1c	

	4.	[illegible] Inspector tra[illegible]		
	5.	Plant Drills		
	6.	Long-term Upgrading of Training and Qualifications	Id/Ic	Refer to [illegible]
	7.	Accreditation of Training Institutions	Ib/Ic	
I.A.3		Licensing and Requalification of Operating Personnel		
	1.	Revise Scope and Criteria for Licensing Exams	2/Ic	Refer to Action Plan Item I.A.3.2
	2.	Operator Licensing Program Changes	Ib/Ic	
	3.	Requirements for Operator Fitness	Ib/Ic	
	4.	Licensing of Additional Operations Personnel	Ib/Ic	
	5.	Establish Statement of Understanding with INPO and DOE	Ib/Ic	
I.A.4		Simulator Use and Development		
	1.	Initial Simulator Improvement	Id/Ic	Refer to Action Plan Item I.A.4.2

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Long-Term Training Simulator Upgrade	4/1c	Refer to Appendix C
	3. Feasibility Study of Procurement of NRC Training Simulator	1b/1b	
	4. Feasibility Study of NRC Engineering Computer	1b/1b	
I.B	Support Personnel		
I.B.1	Management for Operations		
	1. Organization and Management Long-Term Improvements	2/1a	
	2. Evaluation of Organization and Management Improvements of NIOL Applicants	1d/1c	Refer to Action Plan Item I.B.1.1
	3. Loss of Safety Function	1b/1c	
I.B.2	Inspection of Operating Reactors		
	1. Revise IE Inspection Program	1b/1c	
	2. Resident Inspector at Operating Reactors	1b/1b	
	3. Regional Evaluations	1b/1b	
	4. Overview of Licensee Performance	1b/1b	

TABLE I (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
I.C	Operating Procedures		
	1. Short-Term Accident Analysis and Procedures Revision	2/1c	
	2. Shift and Relief Turnover Procedures	2/1c	
	3. Shift Supervisor Responsibilities	2/1c	
	4. Control Room Access	2/1c	
	5. Procedures for Feedback of Operating Experience	5/5	Refer to Appendix C
	6. Procedures for Verification of Correct Performance of Operating Activities	2/1c	
	7. NSSS Vendor Review of Procedures	2/1c	
	8. Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants	1b/1c	
	9. Long-Term Program Plan for Upgrading of Procedures	4/1c	Refer to Appendix C
I.D	Control Room Design		
	1. Control Room Design Reviews	4/4	Refer to Appendix C
	2. Plant Safety Parameter Display Console	4/4	Refer to Appendix C

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	3. Safety System Status Monitoring	4/4	Refer to Appendix C
	4. Control Room Design Standard	lb/lb	
	5. Improved Control Room Instrumentation Research	lb/lb	
	6. Technology Transfer Conference	lb/lb	
I.E	Analysis and Dissemination of Operating Experience		
	1. Office for Analysis and Evaluation of Operation Data	lb/lb	
	2. Program Office Operational Data Activities	lb/lb	
	3. Operational Safety Data Analysis	lb/lb	
	4. Coordination of Licensee, Industry, and Regulatory Programs	4/4	Refer to Appendix C
	5. Nuclear Plant Reliability Data System	lb/lb	
	6. Reporting Requirements	lb/lb	
	7. Foreign Sources	lb/lb	
	8. Human Error Rate Analysis	lb/lb	

TABLE I (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
I.F	Quality Assurance		
	1. Expand QA List	5/5	Refer to Appendix C
	2. Develop More Detailed QA Criteria	5/5	Refer to Appendix C
I.G	Preoperational and Low-Power Testing		
	1. Training Requirements	1b/1c	
	2. Scope of Test Program	1b/1c	
II.	Siting and Design		
II.A	Siting		
	1. Siting Policy Reformulation	1b/1c	
	2. Site Evaluation of Existing Facilities	4/1c	Refer to Appendix C
II.B	Consideration of Degraded or Melted Cores in Safety Review		
	1. Reactor Coolant System Vents	4/4	Refer to Appendix C
	2. Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-accident Operation	4/4	Refer to Appendix C
	3. Post-accident Sampling	4/4	Refer to Appendix C

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	4. Training for Mitigating Core Damage	2/1c	
	5. Research on Phenomena Associated with Core Degradation and Fuel Melting	1b/1b	
	6. Risk Reduction for Operating Reactors at Sites with High Population Densities	1c/1c	
	7. Analysis of Hydrogen Control	1d/1d	Refer to Action Plan Item II.B.8
	8. Rulemaking Proceeding on Degraded Core Accidents	5/5	Refer to Appendix C
II.C	Reliability Engineering and Risk Assessment		
	1. Interim Reliability Evaluation Program (IREP)	1b/1b	
	2. Continuation of IREP	1b/1b	
	3. Systems Interaction	1c/1c	
	4. Reliability Engineering	3/3	Refer to Appendix C
II.D	Reactor Coolant System Relief and Safety Valves		
	1. Testing Requirements	4/4	Refer to Appendix C

TABLE I (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/MI	COMMENTS
	2. Research on Relief and Safety Valve Test Requirements	1b/1b	
	3. Relief and Safety Valve Position Indication	4/4	Refer to Appendix C
II.E	System Design		
II.E.1	Auxiliary Feedwater System		
	1. Auxiliary Feedwater System Evaluation	3/3	Refer to Appendix C
	2. Auxiliary Feedwater System Automatic Initiation and Flow Indication	4/4	Refer to Appendix C
	3. Update Standard Review Plan and Develop Regulatory Guide	1b/1b	
II.E.2	Emergency Core Cooling System		
	1. Reliance on ECCS	2/1b	
	2. Research on Small Break LOCAs and Anomalous Transients	1b/1b	
	3. Uncertainties in Performance Predictions	1b/1b	

TABLE I (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/MI	COMMENTS
II.E.3	Decay Heat Removal		
	1. Reliability of Power Supplies for Natural Circulation	4/4	Refer to Appendix C
	2. Systems Reliability	1b/1b	
	3. Coordinated Study of Shutdown Heat Removal	1d/1d	Refer to Action Plan II.C.4
	4. Alternate Concepts Research	1b/1b	
	5. Regulatory Guide	1b/1b	
II.E.4	Containment Design		
	1. Dedicated Penetrations	5/5	Refer to Appendix C
	2. Isolation Dependability	4/4	Refer to Appendix C
	3. Integrity Check	2/1c	
	4. Purging	4/4	Refer to Appendix C
II.E.5	Design Sensitivity of B&W Reactors		
	1. Design Evaluation	4/1c	Refer to Appendix C
	2. B&W Reactor Transient Response Task Force	4/1c	Refer to Appendix C

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
II.E.6	In-situ Testing of Valves		
	1. Test Adequacy Study	1b/1b	
II.F	Instrumentation and Control		
	1. Additional Accident Monitoring Instrumentation	4/4	Refer to Appendix C
	2. Identification of and Recovery from Conditions leading to Inadequate Core Cooling	4/4	Refer to Appendix C
	3. Instrumentation for Monitoring Accident Conditions (Reg. Guide 1.97)	4/4	Refer to Appendix C
	4. Study of Control and Protective Action Design Requirements	1b/1b	
	5. Classification of Instrumentation, control and Electric Equipment	1b/1b	
II.G	Electrical Power		
	1. Power Supplies for Pressurized Relief Valves, Block Valves, and Level Indicators	4/4	Refer to Appendix C
II.H	TMI-2 Cleanup and Examination		
	1. Maintain Safety of TMI-2 and Minimize Environmental Impact	1b/1b	

TABLE I (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Obtain Technical Data on the Conditions Inside the IMI-2 Containment Structure	lb/lb	
	3. Evaluate and Feedback Information Obtained from IMI	lb/lb	
	4. Determine Impact of IMI on Socioeconomic and Real Property Values	lb/lb	
II.J	General Implications of IMI for Design and Construction Activities		
II.J.1	Vendor Inspection Program		
	1. Establish a Priority System for Conducting Vendor Inspections	lb/lb	
	2. Modify Existing Vendor Inspection Program	lb/lb	
	3. Increase Regulatory Control Over Present Non-licenses	lb/lb	
	4. Assign Residents Inspectors to Vendors and Architect-Engineers	lb/lb	
II.J.2	Construction Inspection Program		
	1. Reorient Inspection Program	lb/lb	

TABLE I (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Increase Emphasis on Independent Measurement in the Construction Inspection Program	lb/lb	
	3. Assign Resident Inspectors to all Construction Sites	lb/lb	
II.J.3	Management for Design and Construction		
	1. Organization and Staffing to Oversee Design and Construction	5/5	Refer to Appendix C
	2. Issue Regulatory Guide	lb/lb	
II.J.4	Revise Deficiency Reporting Requirements		
	1. Revise Deficiency Reporting Requirements	lb/lb	
II.K	Measures to Mitigate Small-Break LOCAs and Loss of Feedwater Accidents		
	1. IE Bulletins	See Table C.1	
	2. Commission Orders on B&W plants	See Table C.2	
	3. Final Recommendations of B&O Task Force	See Table C.3	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
III.	Emergency Preparations and Radiation Effects		
III.A	NRC and Licensee Preparedness		
III.A.1	Improve Licensee Emergency Preparedness - Short-term		
	1. Upgrade Emergency Preparedness	1b/1b	
	2. Upgrade Licensee Emergency Support Facilities	4/4	Refer to Appendix C; NUREG-0654, Rev. 1; and NUREG-0696 for definitive criteria.
	3. Maintain Supplies of Thyroid Blocking Agent (Potassium Iodide)	2/1c	Refer to NUREG-0654, Rev. 1; Stockpiling for the general public is under consideration
III.A.2	Improving Licensee Emergency Preparedness - Long-term		
	1. Amend 10 CFR 50 and 10 CFR 50, Appendix E	4/4	Refer to Appendix C
	2. Development of Guidance and Criteria	1e/1b	
III.A.3	Improving NRC Emergency Preparedness		
	1. NRC Role in Responding to Nuclear Emergencies	1b/1b	
	2. Improve Operations Centers	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/MI	COMMENTS
	3. Communications	4/1c	Refer to Appendix C
	4. Nuclear Data Link	1b/1b	
	5. Training, Drills, and Tests	2/1c	
	6. Interaction of NRC with Other Agencies	1b/1b	
III.B	Emergency Preparedness of State and Local Governments		
	1. Transfer of Responsibilities to FEMA	1b/1b	
	2. Implementation of NRC's and FEMA's Responsibilities	1b/1b	
III.C	Public Information		
	1. Have Information Available for the News Media and the Public	1b/1b	
	2. The Office of Public Affairs will Develop Agency Policy and Provide Training for Interfacing with the News Media and Other Interested Parties	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
III.D	Radiation Protection		
III.D.1	Radiation Source Control		
	1. Primary Coolant Sources Outside the Containment Structure	4/4	Refer to Appendix C
	2. Radioactive Gas Management	1b/1b	
	3. Ventilation System and Radioiodine Adsorber Criteria	1b/1b	
	4. Radwaste System Design Features to Aid in Accident Recovery and Decontamination	1b/1b	
III.D.2	Public Radiation Protection Improvement		
	1. Radiological Monitoring of Effluents	1b/1b	
	2. Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis	1b/1b	
	3. Liquid Pathway Radiological Control	2/1b	
	4. Offsite Dose Measurements	2/1c	
	5. Offsite Dose Calculation Manual	2/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	6. Independent Radiological Measurements	1b/1b	
III.D.3	Worker Radiation Protection Improvements		
	1. Radiation Protection Plans	4/4	Refer to Appendix C
	2. Health Physics Improvements	1b/1b	
	3. Inplant Radiation Monitoring	4/4	Refer to Appendix C
	4. Control Room Habitability	4/4	Refer to Appendix C
	5. Radiation Worker Exposure Data Base	1a/1c	
IV.	Practices and Procedures		
IV.A	Strengthen Enforcement Process		
	1. Seek Legislative Authority	1b/1b	
	2. Revise Enforcement Policy	1b/1b	
IV.B	Issuance of Instructions and Information to Licensees		
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
IV.C	Extend Lessons Learned to Licensed Activities Other than Power Reactors		
IV.C.1	Extend Lessons Learned from TMI to other NRC Programs	1b/1b	
IV.D	NRC Staff Training		
IV.D.1	NRC Staff Training	1b/1b	
IV.E	Safety Decision-Making		
	1. Expand Research on Quantification of Safety Decision-Making	1b/1b	
	2. Plan for Early Resolution of Safety Issues	1b/1b	
	3. Plan for Resolving Issues at Construction Permit Stage	1b/1b	
	4. Resolve Generic Issues by Rulemaking	1b/1b	
	5. Assess Currently Operating Reactors	1b/1b	
IV.F	Financial Disincentive to Safety		
	1. Increased IE Scrutiny of Power Ascension Test Program	1b/1b	

TABLE 1 (Continued)

	ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
	2. Evaluate the Impact of Financial Disincentives to the Safety of Nuclear Power Plants	1b/1b	
IV.G	Improve Safety Rulemaking Procedures		
	1. Develop a Public Agenda for Rulemaking	1b/1b	
	2. Periodic and Systematic Reevaluation of Existing Rules	1b/1b	
	3. Improve Rulemaking Procedures	1b/1b	
	4. Study Alternative for Improved Rulemaking Process	1b/1b	
IV.H	NRC Participation in the Radiation Policy Council		
V.	NRC Policy, Organization and Management		
	1. Develop NRC Policy Statement on Safety	1b/1b	
	2. Study Elimination on Non-safety Responsibilities	1b/1b	
	3. Strengthen Role of ACRS	1b/1b	
	4. Study Need for Additional Advisory Committees	1b/1b	

TABLE 1 (Continued)

ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
5. Improve Public and Intervenor Participation in Hearing Process	1b/1b	
6. Study Construction-During-Adjudication Rules	1b/1b	
7. Study Need for TMI-Related Legislation	1b/1b	
8. Study the Need to Establish an Independent Nuclear Safety Board	1b/1b	
9. Study the Reform of the Licensing Process	1b/1b	
10. Study NRC Top Management Structure and Process	1b/1b	
11. Reexamine Organization and Functions of NRC Offices	1b/1b	
12. Revise Delegations of Authority to Staff	1b/1b	
13. Clarify and Strengthen the Respective Roles of Chairman, Commission, and EDO	1b/1b	
14. Authority to Delegate Emergency Response Functions to a Single Commissioner	1b/1b	

TABLE 1 (Continued)

ACTION ITEM	REQUIREMENT CATEGORY ASSIGNMENT CP/ML	COMMENTS
15. Achieve Single Location - long-term	1b/1b	
16. Achieve Single Location - Interim	1b/2b	
17. Reexamine Commission Role in Adjudication	1b/1b	

TABLE C.1 OFFICE OF INSPECTION AND ENFORCEMENT BULLETINS

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
1.	Review IMI-2 PNs and detailed chronology of IMI-2 accident.	79-50&-5A (Item 1) 79-06&06A (Item 1) 79-06&06B (Item 1)	BWR and PWR	Id/Id	Refer to Action Plan Items I.A.2.2 and I.A.3.1
2.	Review transients similar to IMI-2 that have occurred at other facilities and NRC evaluation of Davis-Besse transient.	79-05&05A (Item 2)	B&W	Id/Ic	Refer to Action Plan Items I.A.2.2 and I.A.3.1
3.	Review operating procedures for recognizing, preventing, and mitigating void formation in transients and accidents	79-05&05A (Item 3) 79-06&06A (Item 2) 79-06&06B (Item 2)	PWR	Id/Ic	Refer to Action Plan Item I.C.1
4.	Review operating procedures and training instructions to ensure that: a. Operators to not override ESF actions unless continued operation is unsafe;	79-05&05A (Item 4.a) 79-05B (Item 2) 79-06A (Item 7.a) 79-06B (Item 6.a) 79-08 (Item 5.a)	PWR and BWR	Id/Ic	Refer to Action Plan Items I.C.1, I.C.7, I.C.8, and I.G.1
	b. HPI system operation	NUREG-0645 (App. G) NUREG-0565 (Rec. 104) 69-110 6002-00 (11/1/79) 69-110 6003-00 (11/20/79) 69-110 6001-00 (11/1/79)	W,CE B&W ANO-1 Davis-Besse 1 Oconee 1, 2 & 3 Crystal River 3 Rancho Seco 1	Id/Ic	Refer to Action Plan Item I.C.1

TABLE C.1 (Continued)

Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
c. RCP operation	NUREG-0623	PWR	Id/Ic	Refer to Action Plan Items I.A.1.3 and I.C.1
d. Operators are instructed not to rely on level indication alone in evaluating plant conditions.	79-05A (Item 4.d) 79-06A (Item 7.d) 79-06B (Item 6.d) 79-08 (Item 5.b)	PWR and BWR	Id/Ic	Refer to Action Plan Items I.C.1, I.A.3.1, and II.F.2
5. Safety-related valve position	79-05&05A (Item 5)	PWR and BWR	Id/Id	Refer to Action Plan Items I.C.2 and I.C.6
a. Review all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper ESF functioning, if required.	79-06A (Item 8) 79-06B (Item 7) 79-008 (Item 6)			
b. Verify that AFW valves are in open position. See Requirement 8 below	79-05A (Item 5)	B&W	Id/Id	Refer to Action Plan Items I.C.2 and I.C.6
6. Review containment isolation initiation design and procedures. Assure isolation of all lines that do not degrade safety features or cooling capability upon automatic initiation of SI.	79-05A (Item 6) 79-06A (Item 4) 79-06B (Item 3) 79-08 (Item 2)	PWR and BWR	Id/Id	Refer to Action Plan Item II.E.4.2
7. Implement positive position controls on valves that could compromise or defeat AFW flow.	79-05A (Item 7)	B&W	Id/Ic	Refer to Action Plan Item II.E.1.1

TABLE C.1 (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
8.	Immediately implement procedures that assure two independent 100% AFW flow paths, or specify explicitly ICO with reduced AFW capacity.	79-05A (Item 8)	B&W	Id/Ic	Refer to Action Plan Item II.E.1.1
9.	Review procedures to assure that radioactive liquids and gases are not transferred out of containment inadvertently (especially upon ESF reset). List all applicable systems and interlocks.	79-05A (Item 9) 79-06A (Item 9) 79-06B (Item 8) 79-08 (Item 7)	PWR and BWR	Id/Ic	Refer to Action Plan Item II.E.4.2
10.	Review and modify (as required) procedures for removing safety-related systems from service (and restori ; to service) to assure operability status is known.	79-05A (Item 10) 79-06A (Item 10) 79-06B (Item 9) 79-08 (Item 8)	PWR and BWR	Id/Ic	Refer to Action Plan Items I.C.2 and I.C.6
11.	Make all operating and maintenance personnel aware of the seriousness and consequences of the erroneous actions taken leading up to, and in early phases of, the TMI-2 accident.	79-05A (Item 11) 79-06A(Item 1.a) 79-06B (Item 1.a)	PWR and BWR	Id/Ic	Refer to Action Plan Items I.A.3.a and I.A.2.2
12.	One hour notification requirement, and continuous communications channel.	79-05B (Item 6) 79-06A (Item 11) 79-06B (Item 10) 79-08 (Item 9)	PWR and BWR	Id/Ic	Refer to Action Plan Items I.E.6 and III.A.3.3

TABLE C.1 (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/Mi Requirement Category Assignment	Comments
13.	Propose Technical Specification changes reflecting implementation of all Bulletin items, as required.	79-05B (Item 7) 79-06A & Rev. 1 (Item 13) 79-06B (Item 12) 79-08 (Item 11)	PWR and BWR	1a/1c	
14.	Review operating modes and procedures to deal with significant amounts of hydrogen.	79-06A (Item 12) 79-06B (Item 11) 79-08 (Item 10)	W, CE GE	1d/1c	Refer to Action Plan Items II.B.4, II.B.7, II.E.4.1 and II.F.1
15.	For facilities with non-automatic AFW initiation, provide dedicated operator in continuous communication with CR to operate AFW.	79-06A (Item 5) 79-06B (Item 4)	W & CE	1d/1c	Refer to Action Plan Item II.E.1.2
16.	Implement (immediately) procedures that identify PRZ PORV "Open" indications and that direct operator to close manually at "RESET" setpoint.	79-06A (Item 6) 79-06B (Item 5)	W & CE	1d/1c	Refer to Action Plan Items I.C.1 and II.D.3
17.	Trip PZR level Bistable so that PZR lo Press. (rather than PZR lo Press. and PZR lo level coincidence) will initiate safety injection. For test, reset lo level bistable.	79-06A & Rev. 1 (Item 3)	W	1c/1c	
18.	Develop procedures and train operators on methods of establishing and maintaining natural circulation	79-05B (Item 1)	B&W	1d/1c	Refer to Action Plan Items I.C.1 and I.G.1

TABLE C.1 (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
19.	Describe design an procedure modifications (based on analysis) to reduce likelihood of automatic P/R POaV actuation in transients.	79-05B (Item 3)	B&W	Id/Ic	Refer to Action Plan Item II.E.5
20.	Provide procedures and training to operators for prompt manual reactor trip for LOFW, IT, MSIV closure, LOOP, LOSG level, & lo P/R level.	79-05B (Item 4)	B&W	2/Ic	
21.	Provide automatic safety-grade anticipatory reactor trip for LOFW, IT, or significant decrease in SG level.	79-05B (Item 5)	B&W	Id/Ic	Refer to Action Plan Item II.K.2.10
22.	Describe automatic and manual actions for proper functioning of auxiliary heat removal systems when FW system not operable.	79-08 (Items 3)	BWR	4/Ic	Refer to Appendix C
23.	Describe uses and types of RV level indication for automatic and manual initiation safety systems. Also, describe alternative instrumentation.	79-08 (Item 4)	BWR	4/Ic	Refer to Appendix C
24.	Perform LOCA analyses for a range of small-break sizes and a range of time lapses between reactor trip and RCP trip.	79-05C (short-term Item 2) 79-06C (short-term Item 2)	PWR	Id/Ic	Refer to Action Plan Item I.C.1

TABLE C.1 (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/ML Requirement Category Assignment	Comments
25.	Develop operator action guidelines (based on analyses in Requirement 24 above).	79-05C (short-term Item 3) 79-06C (short-term Item 3)	PWR	Id/Id	Refer to Action Plan Item I.C.1
26.	Revise emergency procedures and train ROs and SROs based on guidelines developed in Requirement 25 above.	79-05C (short-term Item 4) 79-06C (short-term Item 4)	PWR	Id/Ic	Refer to Action Plan I.C.1, I.A.3.a, and I.G.1
27.	Provide analyses and develop guidelines and procedures for inadequate core cooling conditions. Also, define RCP restart criteria.	79-05C (short-term Item 5) 79-06C (short-term Item 5)	PWR	Id/Id	Refer to Action Plan Items I.C.1 and II.F.2
28.	Provide design that will assure automatic RCP trip for all circumstances where required.	NUREG-0623	PWR	Id/Id	Refer to Action Plan Item II.K.3.5

TABLE C.2 REQUIREMENTS FOR NEW B&W PLANTS DERIVED FROM COMMISSION ORDERS ON OPERATING B&W PLANTS

	Requirement	Source for Operating Reactors	Applicability	CP/MI Requirement Category Assignment	Comments
1.	Upgrade timeliness and reliability of AFW system.	Commission Order	B&W	Id/Ic	Refer to Action Plan Item II.E.1
2.	Procedures and training to initiate and control AFW independent of integrated control system.	Commission Order	B&W	Id/Ic	Refer to Appendix C
3.	Hard-wired control-grade anticipatory reactor trips.	Commission Order	B&W	Id/Ic	Refer to Action Plan Item II.K.2.10
4.	Small-break LOCA analysis, procedures and operator training.	Commission Order	B&W	Id/Ic	Refer to Action Plan Items I.A.3.1 and I.C.1
5.	Complete TMI-2 simulator training for all operators.	Commission Order	B&W	Id/Ic	Refer to Action Plan Item I.A.2.6
6.	Reevaluate analysis for dual-level setpoint control.	Commission Order	Davis-Besse 1	Ic/Ic	
7.	Reevaluate transient of September 24, 1977.	Commission Order	Davis-Besse 1	Ic/Ic	
8.	Continued upgrading of AFW system.	Commission Order	B&W	Id/Ic	Refer to Action Plan Item II.E.1

TABLE C.2 (Continued)

	Requirement	Source for Operating Reactors	Applicability	CP/MI Requirement Category Assignment	Comments
9.	Analysis and upgrading of integrated control system.	Commission Order	B&W	4/1c	Refer to Appendix C
10.	Hard-wired safety-grade anticipatory reactor trips.	Commission Order	B&W	4/1c	Refer to Appendix C
11.	Operator training and drilling.	Commission Order	B&W	1d/1c	Refer to Action Plan Items I.A.3.1, I.A.2.2 I.A.2.5, and I.G.a
12.	Transient analysis and procedures for management of small breaks.	Commission Order	B&W	1d/1c	Refer to Action Plan Item I.C.1
13.	Thermal-mechanical report -- effect of HPI on vessel integrity for small-break LOCA with no AFW.	letter, D. Ross to B&W operating plants, 8/121/79	B&W	1b/1c	Refer to Appendix C
14.	Demonstrate that predicted lift frequency of PORVs and SVs is acceptable.	letter, D. Ross to B&W operating plants, 8/21/79	B&W	3/1c	Refer to Appendix C
15.	Analysis of effects of slug flow on once-through steam generator tubes after primary system voiding.	letter, D. Ross to B&W operating-plants, 8/21/79	B&W	1b/1c	Refer to Appendix C

TABLE C.3 FINAL RECOMMENDATIONS OF BULLETINS AND ORDERS TASK FORCE

	Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
1.	Install automatic PORV isolation system and perform operational test.	NUREG-0565 (2.1.2.1) NUREG-0611 (3.2.4.e) 3.2.4.f) NUREG-0635 (3.2.4.a) (3.2.4.b)	PWR	1d/1c	Refer to Action Plan Item II.K.3.2
2.	Report on overall safety effect of PORV isolation system.	NUREG-0565 (2.1.2.d) NUREG-0611 (3.2.4.g) (3.2.4.i) NUREG-0635 (3.2.4.c)	PWRs	3/3	Refer to Appendix C
3.	Report safety and relief valve failures promptly and challenges annually.	NUREG-0565 (2.1.2.c, 2.1.2.e) NUREG-0611 (3.2.4.h) NUREG-0626 (B.14) NUREG-0635 (3.2.4.d)	All	2/2	
4.	Review and upgrade reliability and redundancy of nonsafety equipment for small-break LOCA mitigation	NUREG-0565 (2.3.2.b) NUREG-0611 (3.2.2.b) NUREG-0626 (B.12) NUREG-0635 (3.2.2.b)	All	1b/1b	Refer to Action Plan Items II.C.1, II.C.2, and II.C.3
5.	Continue to study need for C.1.4.c and need for automatic trip of RCPs, then modify procedures or designs as appropriate.	NUREG-0565 (2.3.2.a) NUREG-0611 (3.2.2.a) NUREG-0635 (3.2.2.a) NUREG-0623	PWR	1b/1b	

TABLE C.3 (Continued)

	Requirement	Source	Applicability	CP/MI Requirement Category Assignment	Comments
6.	Instrumentation to verify natural circulation.	NUREG-0565 (2.6.2.b) NUREG-0611 (3.2.3.b) NUREG-0635 (3.2.3.b)	PWR	Id/Id	Refer to Action Plan Item I.C.1, II.F.2, II.F.3
7.	Evaluation of PORV opening probability during overpressure transient.	NUREG-0565 (2.1.2.b)	B&W	Id/Ic	Refer to Action Plan Item II.K.2.14
8.	Further staff consideration of need for diverse decay heat removal method independent of SGs	NUREG-0565 (2.5.2.a) NUREG-0635 (4.2.5), App. VIII) NUREG-0611 (4.2.5, App. VIII)	PWR	Id/Id	Refer to Action Plan Item II.C.1 and II.E.3.3
9.	Proportional integral derivative controller modification.	NUREG-0611 (3.2.4.b)	W	Ic/2	
10.	Anticipatory trip modification proposed by some licensees to confine range of use to high power levels.	NUREG-0611 (3.2.4.c)	W	Ic/2	
11.	Control use of PORV supplied by Control Components Inc., until further review complete.	NUREG-0611 (3.2.4.d)	All	4/4	Refer to Appendix C
12.	Confirm existence of anticipatory trip upon turbine trip.	NUREG-0611 (3.2.4.a)	W	Ic/2	

TABLE C.3 (Continued)

	Requirement	Source	Applicability	CP/MI Requirement Category Assignment	Comments
13.	Separation of HPCI and RCIC system initiation levels. Analysis and implementation.	NUREG-0626 (A.1)	GE	3/1c	Refer to Appendix C
14.	Isolation of isolation condensers on high radiation.	NUREG-0626 (A.2)	GE plants with isolation condenser	1c/1c	
15.	Modify break detection logic to prevent spurious isolation of HPCI and RCIC systems.	NUREG-0626 (A.3)	GE	2/1c	
16.	Reduction of challenges and failures of relief valves - feasibility study and system modification.	NUREG-0626 (A.4)	GE	3/1c	Refer to Appendix C
17.	Report on outage of ECC systems - licensee report and proposed technical specification changes.	NUREG-0626 (A.6)	GE	1a/1c	
18.	Modification of ADS logic - feasibility study and modification for increased diversity for some event sequences.	NUREG-0626 (A.7)	GE	3/1c	Refer to Appendix A

TABLE C.3 (Continued)

	Requirement	Source	Applicability	CP/MI Requirement Category Assignment	Comments
19.	Interlock on recirculation pump loops.	NUREG-0626 (A.8)	GE Non-Jet Pump ORs	Ic/Ic	
20.	Loss of service water for Big Rock Point.	NUREG-0626 (A.9)	Big Rock Point	Ic/Ic	
21.	Restart of core spray and LPCI systems on low level - design and modification.	NUREG-0626 (A.10)	GE	3/Ic	Refer to Appendix C
22.	Automatic switchover of RCIC system suction - verify procedures and modify design.	NUREG-0626 (B.1)	GE	Ic/Ic	
23	Central water level recording.	NUREG-0626 (B.2)	GE	4/Ic	Refer to Appendix C
24.	Confirm adequacy of space cooling for HPCI and RCIC systems.	NUREG-0626 (B.3)	GE	3/Ic	Refer to Appendix C
25.	Effect of loss of AC power on pump seals.	NUREG-0626 (B.4)	GE	3/Ic	Refer to Appendix C

TABLE C.3 (Continued)

	Requirement	Source	Applicability	CP/MI Requirement Category Assignment	Comments
26.	Study effect on RHR reliability of its use for fuel pool cooling.	NUREG-0626 (B.5)	GE	1d/1c	Refer to Action Plan Item II.E.2.1
27.	Provide common reference level for vessel level instrumentation.	NUREG-0626 (B.6)	GE	2/1c	
28.	Study and verify qualification of accumulators on ADS valves.	NUREG-0626 (B.7)	GE	3/1c	Refer to Appendix C
29.	Study to demonstrate performance of isolation condensers with noncondensibles.	NUREG-0626 (B.13)	GE Isolation Condenser ORs	1c/1c	
30.	Revised small-break LOCA methods to show compliance with 10 CFR 50, Appendix K.	NUREG-0565 (2.2.2.a) NUREG-0611 (3.2.1.a) NUREG-0626 (A.12) NUREG-0635 (3.2.1.a) (3.2.5.a)	All	1b/1c	
31.	Plant-specific calculations to show compliance with 10 CFR 50.46.	NUREG-0565 (2.2.2.b) NUREG-0611 (3.2.1.b) NUREG-0626 (A.13, B.10) NUREG-0635 (3.2.a.b)	All	1b/1c	

TABLE C.3 (Continued)

	Requirement	Source	Applicability	CP/MI Requirement Category Assignment	Comments
32.	Provide experimental verification of two-phase natural circulation models.	NUREG-0565 (2.6.2.a) NUREG-0611 (3.2.3.a) NUREG-0635 (3.2.3.a)	PWR	IIb/IIb	Refer to Action Plan Item II.E.2.2
33.	Evaluate elimination of PORV function.	NUREG-0565 (3.5) NUREG-0611 (3.2.4.k) NUREG-0635 (3.2.4.e)	PWR	IIb/IIb	Refer to Action Plan Item II.C.1
34.	RELAP-4 model development.	NUREG-0611 (3.2.5) NUREG-0635 (3.2.5)	PWR	IIb/IIb	Refer to Action Plan Item II.E.2.2
35.	Evaluation of effects of core flood tank injection on small-break LOCAs.	NUREG-0565 (2.2.2.c)	B&W	Id/Ic	Refer to Action Plan Item I.C.1
36.	Additional staff audit calculations of B&W small-break LOCA analyses.	NUREG-0565 (2.4.2.a)	B&W	IIb/Ic	Refer to Action Plan Item I.C.1
37.	Analysis of B&W plant response to isolated small-break LOCA.	NUREG-0565 (2.6.2.c)	B&W	Id/Ic	Refer to Action Plan I.C.1
38.	Analysis of plant response to a small-break LOCA in the pressurizer spray line.	NUREG-0565 (2.6.2.d)	B&W	Id/Ic	Refer to Action Plan Item I.C.1

TABLE C.3 (Continued)

	Requirement	Source	Applicability	CP/MI Requirement Category Assignment	Comments
39.	Evaluation of effects of water slugs in piping caused by HPI and CFT flows.	NUREG-0565 (2.6.2.e)	B&W	Id/Ic	Refer to Action Plan Item I.C.1
40.	Evaluation of RCP seal damage and leakage during a small-break LOCA.	NUREG-0565 (2.6.2.f)	B&W	Id/Ic	Refer to Action Plan Item II.K.2.16
41.	Submit predictions for LOFT Test L3-6 with RCPs running.	NUREG-0565 (2.6.2.g)	B&W	Id/Ic	Refer to Action Plan Item I.C.1
42.	Submit requested information on the effects of non-condensible gases.	NUREG-0565 (2.6.2.h)	B&W	Id/Ic	Refer to Action Plan Item I.C.1
43.	Evaluation of mechanical effects of slug flow on steam generator tubes.	NUREG-0565 (2.6.2.i)	B&W	Id/Ic	Refer to Action Plan II.K.2.15
44.	Evaluation of anticipated transients with single failure to verify no significant fuel failure.	NUREG-0626 (A.14)	GE	3/Ic	Refer to Appendix C
45.	Evaluate depressurization with other than full ADS.	NUREG-0626 (A.15)	GE	3/Ic	Refer to Appendix C

TABLE C.3 (Continued)

	Requirement	Source	Applicability	CP/MI Requirement Category Assignment	Comments
46.	Response to list of concerns from ACRS consultant.	NUREG-0626 (A.17)	GE	Id/Ic	
47.	Test program for small-break LOCA model verification pretest prediction, test program and model verification.	NUREG-0626 (B.9)	GE	Id/Ic	Refer to Action Plan Items I.C.1, and II.E.2.2
48.	Assess change in safety reliability as result of implementation B&OTF recommendations.	NUREG-0626 (B.15)	GE	Id/Ic	Refer to Action Plan Items II.C.1 and II.C.2
49.	Review of Procedures (NRC).	NUREG-0611 (3.4.1) NUREG-0635 (3.4.1)	W, CE	Ib/Ib	Refer to Action Plan I.C.8 and I.C.9
50.	Review of Procedures (NSSS vendors)	NUREG-0611 (3.4.2) NUREG-0635 (3.4.2)	W, CE	Id/Ic	Refer to Action Plan I.C.7 and I.C.9
51.	Symptom-based emergency procedures.	NURGE-0611 (3.4.3) NUREG-0626 (B.8) NUREG-0635 (3.4.3)	W, CE GE	Id/Ic	Refer to Action Plan Item I.C.9
52.	Operator awareness of revised emergency procedures.	NUREG-0626 (A.11)	GE	Id/Ic	Refer to Action Plan Items I.B.1, I.C.2, and I.C.5

TABLE C.3 (Continued)

	Requirement	Source	Applicability	CP/ML Requirement Category Assignment	Comments
53.	Two operators in control room.	NUREG-0626 (A.16)	GE	Id/Ic	Refer to Action Plan Item I.A.1.3
54.	Simulator upgrade for small-break LOCAs.	NUREG-0565 (2.3.2.c) NUREG-0611 (3.3.1.b) NUREG-0626 (B.11) NUREG-0635 (3.3.1.b)	All	Id/Ic	Refer to Action Plan Item I.A.4.1
55.	Operator monitoring of control board.	NUREG-0611 (3.5.1) NUREG-0635 (3.5.1)	W, CE	Id/Ic	Refer to Action Plan Items I.C.1, I.D.2 and I.D.3
56.	Simulator training requirements.	NUREG-0611 (3.3.1.a) NUREG-0635 (3.3.1.a)	W, CE	Id/Ic	Refer to Action Plan Items I.A.3.1, I.A.3.3, and I.A.2.6
57.	Identify water sources prior to manual activation of ADS.	NUREG-0626 (A.5)	GE	Id/Ic	Refer to Action Plan I.C.1

APPENDIX B

ACTION PLAN ITEMS APPLICABLE TO
PENDING CONSTRUCTION PERMIT AND
MANUFACTURING LICENSE APPLICATIONS

			CP	ML
I.A.1.1	-	Shift Technical Advisor	X	
I.A.1.2	-	Shift Supervisor Administrative Duties	X	
I.A.1.3	-	Shift Manning	X	
I.A.2.5	-	Plant Drills	X	
I.A.3.1	-	Revise Scope and Criteria for Licensing Exams	X	
I.A.4.2	-	Long-Term Training Simulator Upgrade	X	
I.B.1.1	-	Organization and Management Long Term Improvements	X	
I.C.1	-	Short-Term Accident Analysis and Procedures Revision	X	X
I.C.2	-	Shift and Relief Turnover Procedures	X	
I.C.3	-	Shift Supervisor Responsibilities	X	
I.C.4	-	Control Room Access	X	
I.C.5	-	Procedures for Feedback of Operating Experience	X	X
I.C.6	-	Procedures for Verification of Correct Performance of Operating Activities	X	
I.C.7	-	NSSS Vendor Review of Procedures	X	
I.C.9	-	Long-Term Program Plan for Upgrading Procedures	X	
I.D.1	-	Control Room Design Reviews	X	X
I.D.2	-	Plant Safety Parameter Display Console	X	X
I.D.3	-	Safety System Status Monitoring	X	X
I.E.4	-	Coordination of Licensee, Industry, and Regulatory Programs	X	X
I.F.1	-	Expand QA List	X	X

			CP	ML
I.F.2	-	Develop More Detailed QA Criteria	X	X
II.A.2	-	Site Evaluation of Existing Facilities	X	
II.B.1	-	Reactor Coolant System Vents	X	X
II.B.2	-	Plant shielding to Provide Access to Vital Areas and Protect Safety Equipment From Post-Accident Operation	X	X
II.B.3	-	Post-Accident Sampling	X	X
II.B.4	-	Training for Mitigating Core Damage	X	
II.B.8	-	Rulemaking Proceeding on Degraded Core Accidents	X	X
II.C.4	-	Reliability Engineering	X	X
II.D.1	-	Testing Requirements	X	X
II.D.3	-	Relief and Safety Valve Indication	X	X
II.E.1.1	-	Auxiliary Feedwater System Evacuation	X	X
II.E.1.2	-	Auxiliary Feedwater System Automatic Initiation and Flow Indication	X	X
II.E.2.1	-	Reliance on ECCS	X	
II.E.3.1	-	Reliability of Power Supplies for Natural Circulation	X	X
II.E.4.1	-	Dedicated Penetrations	X	X
II.E.4.2	-	Isolation Dependability	X	X
II.E.4.3	-	Integrity Check	X	
II.E.4.4	-	Purging	X	X
II.E.5.1	-	Design Evaluation	X	
II.E.5.2	-	B&W Reactor Transient Response Task Force	X	
II.F.1	-	Additional Accident Monitoring Instrumentation	X	X
II.F.2	-	Identification and Recovery from Conditions Leading to Inadequate Core Cooling	X	X
II.F.3	-	Instrumentation for Monitoring Accident Conditions (Reg. Guide 1.97)	X	X

			CP	ML
II.G.1	-	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	X	X
II.J.3.1	-	Organization and Staffing to Oversee Design and Construction	X	X
II.K.1.20	-	Provide Procedures and Training to Operators for Pompt Manual Reactor Trip for LOFW, TT, MSIV Closure, Loop, LOSG Level, and Low Pressurizer Level	X	
II.K.1.22	-	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems when FW System is not Operable	X	
II.K.1.23	-	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation of Safety Systems. Also Describe Alternative Instrumentation.	X	
II.K.2.9	-	Analysis and Upgrading of Integrated Control System	X	
II.K.2.10	-	Hard-Wired Safety-Grade Anticipatory Reactor Trips	X	
II.K.2.14	-	Demonstrate that Predicted Lift Frequency of PORVs and SVs is Acceptable	X	
II.K.2.16	-	Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite Power	X	
II.K.3.2	-	Report on Overall Safety Effect of PORV Isolation System	X	X
II.K.3.3	-	Report Safety and Relief Valve Failures Promptly and Challenges Annually	X	X
II.K.3.9	-	Proportional Integral Derivative Controller Modification		X
II.K.3.10	-	Anticipatory Trip Modification by Some Licensees to Confine Range of Use to High Power Levels		X
II.K.3.11	-	Control Use of PORV Supplied by Control Components, Inc., Until Further Review is Completed	X	X

			CP	ML
II.K.3.13	-	Separation of HPCI and RCIC System Initiation Levels. Analysis and Implementation	X	
II.K.3.15	-	Modify Break Detection Logic to Prevent Spurious Isolation of HPIC and RCIC Systems	X	
II.K.3.16	-	Reduction of Challenges and Failures of Relief Valves. Feasibility Study and System Modification	X	
II.K.3.18	-	Modification of ADS Logic. Feasibility Study and Modification of Increased Diversity of Some Event Sequences	X	
II.K.3.21	-	Restart of Core Spray and LPCI Systems on Low Level Design and Modification	X	
II.K.3.23	-	Central Water Level Recording	X	
II.K.3.24	-	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	X	
II.K.3.25	-	Effect of Loss of AC Power on Pump Seals	X	
II.K.3.27	-	Provide Common Reference Level for Vessel Level Instrumentation	X	
II.K.3.28	-	Study and Verify Qualification of Accumulators on ADS Valves	X	
II.K.3.44	-	Evalution of Anticipated Transients with Single Failure to Verify no Significant Fuel Failure	X	
II.K.3.45	-	Evaluate Depressurization with Other Than Full ADS	X	
III.A.1.2	-	Upgrade License Emergency Support Facilities	X	X
III.A.2.1	-	Amend 10 CFR 50 and 10 CFT 50, Appendix E	X	X
III.A.3.3	-	Communications	X	
III.A.3.5	-	Training, Drills, and Tests	X	
III.D.1.1	-	Primary Coolant Sources Outside the Containment Structure	X	

			CP	ML
III.D.2.3	-	Liquid Pathway Radiological Control	X	
III.D.2.4	-	Offsite Dose Measurements	X	
III.D.2.5	-	Offsite Dose Calculation Manual	X	
III.D.3.1	-	Radiation Protection Plans	X	X
III.D.3.3	-	In-Plant Radiation Monitoring	X	X
III.D.3.4	-	Control Room Habitability	X	X

APPENDIX C
INFORMATION REQUIREMENTS FOR TMI-2 ACTION PLAN ITEMS IN CATEGORIES 3, 4, AND 5

I.A.4.2 LONG-TERM TRAINING SIMULATOR UPGRADE

Applicants shall describe their program for providing simulator capability for their plants. In addition, they shall describe how they will assure that their proposed simulator will correctly model their control room. Applicants shall provide sufficient information to permit the NRC staff to verify that they will have the necessary simulator capability to carry out the actions described in this Action Plan item as well as Action Plan Item II.K.3.54. Applicants shall submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient details shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING, DESIGN AND CONSTRUCTION EXPERIENCE

Applicants shall submit a description of their administrative procedures for the evaluation of operating, design, and construction experience and describe how they will assure that applicable important industry experiences originating from both within and outside the applicant's construction organization will continually be provided to those designing and constructing the plant. Applicants shall submit a general discussin of how the requirments will be met. These procedures shall: (1) Clearly identify organization responsibilities for review and identification of these important experiences and the feedback of pertinent information to those responsible for designing and constructing the plant; (2) Identify the administrative and technical review steps necessary in implementing applicable important experiences; (3) Identify the recipients of various categories of information from these experiences or otherwise provide means through which such information can be readily related to the job functions of the recipients; (4) Assure that applicant and contractor personnel do not routinely receive extraneous and unimportant experience-related information in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency; (5) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to applicant and contractor personnel for implementation until resolution is reached; and (6) Provide practical interim audits to assure that the feedback program functions effectively at all levels. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of construction permits or manufacturing license.

I.C.9 LONG-TERM PROGRAM PLAN FOR UPGRADING OF PROCEDURES

Applicants shall describe their program plan which is to begin during construction and follow into operation for integrating and expanding current efforts in the area of plant procedures. The scope of the program should include emergency

procedures, reliability analysis, human factors engineering, crisis management and operator training. Applicants shall also insure that their program will be coordinated, to the extent possible, with INPO and other industry group efforts. Applicants will submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

I.D.1 CONTROL ROOM DESIGN REVIEWS

Applicants shall provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Applicants shall provide a general discussion of their approach to control room designs that comply with human factor principles by specifying the design concept selected and the supporting design bases and criteria. Cosmetic revisions to conventional (1960 technology) designs is unacceptable. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. Applicants shall commit to control room designs complying to human factors principles prior to issuance of a CP or ML and shall supply design information for approval prior to committing to fabrication or revision of fabricated control room panels and layouts.

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Applicants shall describe how they intend to meet the staff criteria contained in NUREG-0696 for the plant safety parameter display console. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

I.D.3 SAFETY SYSTEM STATUS MONITORING

Applicants shall describe how their design conforms to Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

I.E.4 COORDINATION OF LICENSEE, INDUSTRY AND REGULATORY PROGRAMS

Applicants shall, in conjunction with Action Plan Item I.C.5, provide a description of their program to evaluate experience both at their own plant and similar plants and factor this experience, as appropriate, into the design and construction of their plant. In addition, the program shall describe how these activities will be factored into the operation of the plant. Applicants shall submit, prior to the issuance of construction permits or manufacturing license, a general discussion of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly.

I.F.1 EXPAND QA LIST

Prior to issuance of the constuction permits or manufacturing license, applicants shall revise their QA programs by expanding their QA lists to include all items and activities affecting safety as defined by Regulatory Guide 1.29 and Appendix A to 10 CFR Part 50, and shall provide a commitment to apply the revised QA program to all such items and activities.

I.F.2 DEVELOP MORE DETAILED QA CRITERIA

Applicants shall describe the changes to their QA programs that have resulted from their review of the accident at TMI-2. In addition, applicants shall address the appropriate matters discussed in this Action Plan item and the extent to which they have been considered in their QA program. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a revised description of their QA program that includes consideration of these matters.

II.A.2 SITE EVALUATION OF EXISTING FACILITIES

"Alternatives under consideration".

II.B.1 REACTOR COOLANT SYSTEM VENTS

Applicants shall (1) modify their plant designs as necessary to provide high point reactor coolant system and reactor vessel head vents that can be remotely operated from the control room. Applicants shall, to the extent possible,

I.E.4 COORDINATION OF LICENSEE, INDUSTRY AND REGULATORY PROGRAMS

Applicants shall, in conjunction with Action Plan Item I.C.5, provide a description of their program to evaluate experience both at their own plant and similar plants and factor this experience, as appropriate, into the design and construction of their plant. In addition, the program shall describe how these activities will be factored into the operation of the plant. Applicants shall submit, prior to the issuance of construction permits or manufacturing license, a general discussion of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly.

I.F.1 EXPAND QA LIST

Prior to issuance of the constuction permits or manufacturing license, applicants shall revise their QA programs by expanding their QA lists to include all items and activities affecting safety as defined by Regulatory Guide 1.29 and Appendix A to 10 CFR Part 50, and shall provide a commitment to apply the revised QA program to all such items and activities.

I.F.2 DEVELOP MORE DETAILED QA CRITERIA

Applicants shall describe the changes to their QA programs that have resulted from their review of the accident at TMI-2. In addition, applicants shall address the appropriate matters discussed in this Action Plan item and the extent to which they have been considered in their QA program. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a revised description of their QA program that includes consideration of these matters.

II.A.2 SITE EVALUATION OF EXISTING FACILITIES

"Alternatives under consideration".

II.B.1 REACTOR COOLANT SYSTEM VENTS

Applicants shall (1) modify their plant designs as necessary to provide high point reactor coolant system and reactor vessel head vents that can be remotely operated from the control room. Applicants shall, to the extent possible,

provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.B.2 PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

Applicants shall (1) perform radiation and shielding design reviews of spaces around systems that may contain highly radioactive fluids and (2) implement plant design modifications necessary to permit adequate access to vital areas and protect safety equipment. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.B.3 POST-ACCIDENT SAMPLING

Applicants shall (1) review the reactor coolant and containment atmosphere sampling system designs and the radiological spectrum and chemical analysis facility designs, and (2) modify their plant designs as necessary to meet the requirements. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

"Alternatives under consideration".

provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.B.2 PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

Applicants shall (1) perform radiation and shielding design reviews of spaces around systems that may contain highly radioactive fluids and (2) implement plant design modifications necessary to permit adequate access to vital areas and protect safety equipment. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

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II.B.8 RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

"Alternatives under consideration".

preventive and mitigative, they propose to include at their facilities that have the potential for significant risk reduction.

II.C.4 RELIABILITY ENGINEERING AND RISK ASSESSMENT

Applicants shall perform simplified system reliability analyses for the following systems: subcriticality systems, emergency feedwater systems (PWRs), reactor core isolation cooling system, (BWRs), emergency core cooling system injection and recirculation systems, shutdown cooling system, containment cooling and spray systems, safety features actuation systems, and auxiliary systems upon which these depend (alternating and direct current, compressed air, essential service water or cooling systems, and heating, ventilating and air conditioning systems). These analyses shall use event-tree and fault-tree logic techniques to identify design weaknesses and possible system modifications that could be made to improve the capability and reliability of the above systems under various transient and loss of coolant accident events. Particular emphasis shall be given to determining potential failures that could result from human errors, common causes, single point vulnerabilities, and test and maintenance outages.

Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.D.1 TESTING REQUIREMENTS

Applicants and their agents shall plan and carry out a test program and model development to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test planning. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a general explanation of how the testing requirements will be met. Sufficient detail should be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Applicants shall (1) demonstrate the applicability of the generic tests conducted under II.D.1 to their particular plants and (2) modify their plant designs as necessary. Applicants shall commit, prior to the issuance of the construction permits or manufacturing license, to comply with these requirements and shall submit within six months following the completion of the generic tests or the issuance of construction permits, whichever is later, a detailed explanation of how the test results will be incorporated in the plant design. Sufficient detail should be presented to provide reasonable assurance that the requirements resulting from the test will be implemented properly prior to the issuance of operating licenses.

II.D.3 RELIEF AND SAFETY VALVE POSITION INDICATION

Applicants shall modify their plant designs as necessary to provide direct indication of relief and safety valve position in the control room. Applicants shall, to the extent possible, provide preliminary design information at a

(Third paragraph - II.C.4 Reliability Engineering and Risk Assessment)

The staff is developing a program, applicable industry-wide, that would require performance of risk assessments for all plants. At such time as this program is implemented, CP holders might be required to perform such assessments. If such risk assessments are required by CP holders, that assessment would replace the requirement to perform the simplified reliability analysis described above.

level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to issuance of operating licenses.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Applicants shall perform a reevaluation of their proposed auxiliary feedwater (AFW) system. This reevaluation shall include the following:

(1) Performance of simplified auxiliary feedwater system reliability analyses using event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss of main feedwater transient conditions, with particular emphasis being given to determining potential failures that could result from human errors, common causes, single point vulnerabilities, and test and maintenance outages. The results of this evaluation shall be compared with the results of the NRC staff's generic AFW system evaluation published in Appendix III to NUREG-0611 and Appendix III to NUREG-0635. Applicants with plants with AFW systems with relatively low reliabilities shall submit proposed design changes and/or procedural actions which will improve the relative reliability of the AFW system to above average. Applicants whose plant designs do not include high head high pressure injection system pumps for use in the feed and bleed mode of decay heat removal in case of AFW system failure shall assure that their AFW system has a very high reliability relative to those AFW systems evaluated by the NRC and staff and reported in NUREG-0611 and NUREG-0635 respectively.

(2) Completion of a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 as principal guidance. This requirement applies to those plants where the Standard Review Plan was not used as criteria during the NRC staff's CP review.

(3) Reevaluation of the AFW system flow design bases and criteria. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.E.1.2. AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

Applicants with PWR plants which have manually initiated auxiliary feedwater (AFW) systems shall submit (1) proposed automatically initiated safety-grade designs which meet the requirements specified in Sections 2.1.7.a and 2.1.7.b of NUREG-0578, and (2) analyses of a potential unreviewed safety issue relating to automatic AFW system initiation with a postulated main steam line break inside containment and its effect on containment pressure design capability and return to reactor power. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting

design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Applicants with PWR plants which have automatically initiated AFW systems shall provide information in sufficient detail to provide reasonable assurance that their designs are safety-grade and meet the requirements specified in Sections 2.1.7.a and 2.1.7.b of NUREG-0578.

II.E.3.1 RELIABILITY OF POWER SUPPLIES FOR NATURAL CIRCULATION

Applicants shall (1) upgrade the power supplies for the pressurizer heaters and associated motive and control power interfaces to meet the applicable requirements specified in Section 2.1.1 of NUREG-0578 and (2) establish procedures and training for maintaining the reactor coolant system at hot standby conditions with only onsite power available.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.E.4.1 DEDICATED PENETRATION

Applicants for plant designs with external hydrogen recombiners shall modify their applications as necessary to include redundant dedicated containment penetrations so that the recombiner systems can be connected to the containment atmosphere without violating single-failure criteria, such as having to open large containment purging ducts or otherwise jeopardizing the containment function. Applicants shall submit, prior to the issuance of construction permits or the manufacturing license, a detailed explanation of how the requirements will be met in order to provide reasonable assurance that the requirements will be implemented properly.

II.E.4.2 ISOLATION DEPENDABILITY

Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4.

All plants shall give careful consideration to the definition of essential and non-essential systems, identify each system determined to be essential, identify each system determined to be non-essential, and describe the basis for selection of each essential system. All non-essential systems shall be automatically isolated by the containment isolation signal. Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus non-essential systems and is due to be issued by June 1981.

For post-accident situations, each non-essential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56, and 57, as clarified by Standard Review Plan, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by Standard Review Plan, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a non-essential penetration must receive diverse isolation signals.

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require delibrate operator action. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting this requirement.

Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

The containment setpoint pressure that initiates containment isolation for non-essential penetrations must be reduced to the minimum compatible with normal operating conditions. The containment pressure history during normal operation for similar operating plants should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any propsed values greater than 1 psi will require detailed justification.

All systems that provide an open path from the containment to the environs (e.g., containment purge and vent systems) must close on a safety-grade high radiation signal.

Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979, must be sealed closed as defined in SRP 6.2.4, item II.3f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit state of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.E.4.4 PURGING

Applicants shall (1) address restricted purging and justification of any unrestricted purging, (2) evaluate the performance of purging and venting isolation valves against accident pressure, (3) address the interim NRC guidance on valve operability and (4) adopt procedures and restrictions consistent with the revised requirements.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.E.5.1 DESIGN EVALUATION

Applicants with B&W-designed reactors shall (1) identify the most severe overcooling events (considering both anticipated transients and accidents) that could occur at the facilities, (2) show, in view of the arrival rate for these events, that the design criterion for the number of actuation cycles of the emergency core cooling system and reactor protection system is adequate, (3) recommend changes to systems or procedures that would reduce primary system sensitivity. Applicants with B&W-designed reactors shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.E.5.2 B&W REACTOR TRANSIENT RESPONSE TASK FORCE

Applicants with B&W-designed reactors shall address the additional licensing requirements resulting from this action plan item when issued. Applicants with B&W-designed reactors shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

Applicants shall comply with the requirements addressed to construction permit applicants in NRR letters dated October 10, 1979 and November 9, 1979. Applicants

shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.F.2 INDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING

Applicants shall describe their program for developing and implementing procedures to be used by the reactor operators to detect and recover from conditions leading to inadequate core cooling.

Applicants with PWR plants shall incorporate in their plant designs a primary coolant saturation meter and all applicants shall incorporate in their plant designs instrumentation to detect conditions with a potential that may lead to inadequate core cooling. Any additional equipment, including reactor water level instrumentation, that could be used to indicate inadequate core cooling shall be incorporated in the plant designs. Design requirements for core exit thermocouples are described in NUREG-0737.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.F.3 INSTRUMENTATION FOR MONITORING ACCIDENT CONDITIONS (REG. GUIDE 1.97)

Applicants shall provide in their facility design instrumentation to monitor plant variables and systems during and following an accident in accordance with defined design bases. Recommendation specified in Regulatory Guide 1.97, Rev. 2, December 1980. Designs are already established for much of the instrumentation that will be required; some of the requirements, however, may involve state-of-the-art designs or designs which have yet to be developed.

Applicants shall, to the extend possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.G.1 POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES, BLOCK VALVES, AND LEVEL INDICATION

Applicants with PWR plants shall upgrade the power supplies for the pressurizer relief valves, block valves, and pressurizer level indicators to meet the applicable requirements specified in Section 2.1.1 of NUREG-0578. Applicants with PWR plants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the support design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.J.3.1 ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION

Applicants shall describe their program for the management oversight of design and construction activities. Specific items to be addressed include: (1) the organizational and management structure which is singularly responsible for the direction of the design and construction of the proposed plant, (2) technical resources which are directed by the utility organization, (3) details of the interaction of design and construction within the utility organization and the manner by which the utility will assure close integration of the architect engineer and nuclear steam supply vendor, (4) proposed procedures for handling the transition to operation, and (5) the degree of top level management oversight and technical control to be exercised by the utility during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

Draft NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources" is the keystone for similar development of guidelines for this task. Therefore, the principal applicable elements of NUREG-0731 shall be used by CP and ML applicants in addressing this task.

Applicants shall submit detailed information in order to provide reasonable assurance that the requirements will be implemented properly prior to issuance of the construction permits or manufacturing license.

II.K.1.22 DESCRIBE AUTOMATIC AND MANUAL ACTIONS FOR PROPER FUNCTIONING OF AUXILIARY HEAT REMOVAL SYSTEMS WHEN FW SYSTEM NOT OPERABLE

Applicants with B&W plants shall address the requirements set forth in action item 3 of IE Bulletin 79-80. A general explanation of how these requirements will be met is required prior to issuance of the construction permits. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly.

II.K.1.23 DESCRIBE USES AND TYPES OF RV LEVEL INDICATION FOR AUTOMATIC AND MANUAL INTERACTION OF SAFETY SYSTEMS. ALSO DESCRIBE ALTERNATIVE INSTRUMENTATION

Applicants with BWR plants shall address the requirements set forth in action item 4 of IE Bulletin 79-08. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design basis and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.K.2.9 ANALYSIS AND UPGRADING OF INTEGRATED CONTROL SYSTEM

Applicants with B&W-designed plants shall address the requirements set forth in the Commission Orders regarding the analysis and upgrading of the integrated control system. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.K.2.10 HARD-WIRED SAFETY-GRADE ANTICIPATORY REACTOR TRIPS

Applicants with B&W-designed plants shall address the requirements set forth in the Commission Orders regarding hard-wired, safety-grade anticipatory reactor trips. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.K.2.14 DEMONSTRATE THAT PREDICTED LIFT FREQUENCY OF PORVs AND SVs IS ACCEPTABLE

Applicants with B&W-designed plants shall address the requirements set forth in the Commission Orders regarding demonstration that the predicted lift frequency of power operated relief valves and safety valves is acceptable. Applicants with B&W-designed plants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.2.16 IMPACT OF RCP SEAL DAMAGE FOLLOWING SMALL-BREAK LOCA WITH LOSS OF OFFSITE POWER

Applicants with B&W-designed plants shall address the requirements set forth in the Commission Orders regarding the impact of reactor coolant pump seal damage following a small break loss-of-coolant accident with loss of offsite power. Applicants with B&W-designed plants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.2 REPORT ON OVERALL SAFETY EFFECT OF PORV ISOLATION SYSTEM

Applicants with PWR plants shall address the requirements set forth in Items 3.2.4.e and 3.2.4.f of NUREG-0611. Applicants with PWR plants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.11 CONTROL USE OF PORV SUPPLIED BY CONTROL COMPONENTS, INC. UNTIL FURTHER REVIEW COMPLETE

Applicants with PWR plants shall address the applicable requirements set forth in Item 3.2.4.d of NUREG-0611. Applicants with PWR plants shall submit, prior to the issuance of construction permits or manufacturing license, a general discussion of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.K.3.13 SEPARATION OF HPCI AND RCIC SYSTEM INITIATION LEVELS - ANALYSIS AND IMPLEMENTATION

Applicants with BWR plants shall address the requirements set forth in Item A.1 of NUREG-0626 as they apply to HPCS and RCIC systems. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.16 REDUCTION OF CHALLENGES AND FAILURES OF RELIEF VALVES - FEASIBILITY STUDY AND SYSTEM MODIFICATION

Applicants with BWR plants shall address the requirements set forth in Item A.4 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.18 MODIFICATION OF ADS LOGIC - FEASIBILITY STUDY AND MODIFICATION FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

Applicants with BWR plants shall address the requirements set forth in Item A.7 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates,

and the program to assure that the results of such studies are factored into the final designs.

II.K.3.21 RESTART OF CORE SPRAY AND LPCI SYSTEMS ON LOW LEVEL - DESIGN AND MODIFICATION

Applicants with BWR plants shall address the requirements set forth in Item A.10 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.23 CENTRAL WATER LEVEL RECORDING

Applicants with BWR plants shall address the requirements set forth in Item B.2 of NUREG-0626. Applicants shall implement design modifications as necessary to meet the requirements. Applicants shall submit, prior to issuance of construction permits, a general explanation of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

II.K.3.24 CONFIRM ADEQUACY OF SPACE COOLING FOR HPCI AND RCIC SYSTEMS

Applicants with BWR plants shall address the HPCI and RCIC systems requirements set forth in Item B.3 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.25 EFFECT OF LOSS OF AC POWER ON PUMP SEALS

Applicants with BWR plants shall address the requirements set forth in Item B.4 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.28 VERIFY QUALIFICATION OF ACCUMULATORS ON ADS VALVES

Applicants with BWR plants shall provide information to assure that the ADS valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation while taking no credit for non-safety related equipment or instrumentation. Air (or nitrogen) leakage through valves must be accounted for to assure that enough inventory of compressed air (or nitrogen) will be available to cycle the ADS valves. Applicants shall commit that these requirements will be met in the final design at the OL stage.

In addressing this item prior to CP issuance, applicants should note that safety analysis reports claim that air (or nitrogen) accumulators for the ADS valves provide sufficient capacity (inventory) to cycle these valves open five times at design pressures. Also, General Electric has stated that the emergency

core cooling systems are designed to withstand a hostile environment and still perform their functions for 100 days following an accident.

II.K.3.44 EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO SIGNIFICANT FUEL FAILURE

Applicants with BWR plants shall address the requirements set forth in Item A.14 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.45 EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

Applicants with BWR plants shall address the requirements set forth in Item A.15 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

III.A.1.2 UPGRADE LICENSEE EMERGENCY SUPPORT FACILITIES

Applicants shall address the requirements for a Technical Support Center, Operational Support Center and the Emergency Operations Facility. Applicants shall provide preliminary design information in accordance with the functional requirements of NUREG-0696 at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

III.A.2.1 AMEND 10 CFR 50 AND 10 CFR 50, APPENDIX E

Applicants shall submit, prior to the issuance of construction permits, a discussion of preliminary plans for coping with emergencies addressing the guidance and criteria described in NUREG-0654, Rev. 1, NUREG-0696, Regulatory Guides 1.23 and 1.97, and the requirements of 10 CFR Part 50, and Appendix E to 10 CFR Part 50 as they apply to construction permit applications. Sufficient detail shall be presented to provide reasonable assurance that the emergency plan, emergency procedures and emergency support facilities will be completed, integrated and implemented properly prior to the issuance of operating licenses.

III.A.3.3 COMMUNICATIONS

Applicants shall include provisions in their designs for prompt notification of, and reliable continuous communications with, the NRC Operations Center in

core cooling systems are designed to withstand a hostile environment and still perform their functions for 100 days following an accident.

II.K.3.44 EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO SIGNIFICANT FUEL FAILURE

Applicants with BWR plants shall address the requirements set forth in Item A.14 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

II.K.3.45 EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

Applicants with BWR plants shall address the requirements set forth in Item A.15 of NUREG-0626. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

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III.A.3.3 COMMUNICATIONS

Applicants shall include provisions in their designs for prompt notification of, and reliable continuous communications with, the NRC Operations Center in

the event of accidents and other emergencies. This will involve installation of dedicated telephone lines and possibly a high-frequency radio backup system.

Applicants will submit prior to the issuance of construction permits a general discussion of how requirements in this regard will be met. Sufficient detail shall be presented to provide reasonable assurance that those requirements will be implemented properly prior to the issuance of operating licenses.

III.D.1.1 PRIMARY COOLANT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE

NRC is studying the need for improved acceptance criteria for systems outside containment that contain (or might contain) radioactive material either during normal operations or following an accident. These studies are to be completed in early 1981, and these matters will be included in the degraded-core rulemaking proceeding.

Applicants shall review the designs of such systems outside containment, and their provisions for leakage control and detection, overpressurization design, discharge points for waste gas venting systems, etc., with the goal of minimizing the possibility of exposure to workers and public during normal operations and in the event of an accident.

In this regard, applicants shall submit, prior to the issuance of construction permits, a general discussion of their approach to minimizing leakage from such systems outside containment, in sufficient detail to provide reasonable assurance that this objective will be met satisfactorily prior to issuance of operating licenses.

III.D.3.1 RADIATION PROTECTION PLANS

Applicants shall address Action Plan requirements regarding radiation protection plans which will keep exposures to workers as low as reasonably achievable during both normal operation and accident conditions, and which would allow plant workers to take effective action to control the course and consequences of an accident. A general explanation of how the requirements will be met is required prior to issuance of the construction permits or the manufacturing license. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly.

III.D.3.3 IN-PLANT RADIATION MONITORING

Applicants shall review their designs to assure that provisions for monitoring inplant radiation and airborne radioactivity are appropriate for a broad range of routine and emergency conditions. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

III.D.3.4 CONTROL ROOM HABITABILITY

Applicants shall review the design of their facilities for conformance to requirements stated in the Action Plan. NRC will consider possible new criteria to preclude control room contamination via potential internal pathways indicated by the TMI-2 experience.

Applicants shall address prior to the issuance of the construction permits or manufacturing license, how they will implement the existing requirements set forth in this Action Plan item. Applicants shall also address the extent to which improvements have been made to prevent control room contamination via pathways not previously considered. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.