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PEBBLE SPRINGS
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NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF:

SUBCOMMITTEE MEETING

on

PEBBLE SPRINGS

Place - Portland, Oregon

Date - 28 October 1977

Pages 1 - 30

Telephone:
(202) 347-3700

ACE - FEDERAL REPORTERS, INC.

Official Reporters

444 North Capitol Street
Washington, D.C. 20001

NATIONWIDE COVERAGE - DAILY

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UNITED STATES NUCLEAR REGULATORY COMMISSION

28 October 1977

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING

ON

PEBBLE SPRINGS

Squires Parlor,
Rodeway Inn,
7101 N. E. 82nd Avenue,
Portland, Oregon.

Friday, October 28, 1977.

The ACRS Subcommittee on Pebble Springs met, pursuant to notice, at 1:00 p.m., Dr. Spencer H. Bush, Chairman, presiding.

BEFORE:

DR. SPENCER H. BUSH, Chairman.
MR. HAROLD ETHERINGTON, Member.
DR. STEPHEN LAWROSKI, Member.
MR. JESSE EBERSOLE, Member.
DR. MILTON S. PLESSET, Member.

C O N T E N T S

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P R O C E E D I N G S

DR. BUSH: This is a subset of a subset of a series of Subcommittee meetings. It will deal specifically with some issues on Pebble Springs.

It's a matter of record that the ACRS has written an [illegible] letter. However there is a series of outstanding issues. And I would call upon, at this time, the NRC staff to give us a status report, and I guess indicate what our next goal is with regard to having an SER so that we can proceed.

Now, Steve, you realize if you delay a little longer you'll be doing me a great favor.

(Laughter)

MR. VARGA: Well, we'll have something to say about that.

DR. BUSH: I'll turn it over to you at this stage. We're going to strive mightily to be done here in our two-hour schedule. And if I make use of the gavel, you'll understand why.

MR. STAHLE: Mr. name is Carl Stahle. I'm the NRC Licensing Project Manager. With me today to assist in this review is Mr. Varga.

This project was reviewed by the Committee in February 1976, and it reviewed the SER plus Supplements No. 1 and 2. No. 1 dealt with a number of outstanding issues at that time, and they were addressed in the SER. Supplement No. 2 dealt exclusively with our evaluation of ECCS. At that time we found the ECCS acceptable.

We did not at that time deal with any of the geological or seismological aspects of the site; the reason being, at that time, that there were a number of questions from the U.S.G.S. that were not answered. And the Committee decided at that point to defer hearings on this matter until a later date.

At a much later date, of course, the importance of the 1872 earthquake was recognized; therefore that accounts for the long delay since February 1976 to the present.

With the belief that the resolution of the 1872 earthquake is imminent, the Staff immediately proceeded with updating its review of the non-seismic issues. This was done with the full cooperation of the applicant in order that we may achieve a goal of presenting to you all of the issues at the next full committee meeting, which at this time is scheduled for January, on the basis that we will have the 1872 earthquake in hand.

So with this goal in mind of being able to address all the issues applicable to Pebble Springs, the Staff then proceeded to address the issues that were in existence in February 1976, new issues that came up, as well as proceeding to re-evaluate the nuclear steam supply system in view of the

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current reviews that had been going on with respect to the Babcock and Wilcox standard plant and the current application using the 205 plants.

We have, therefore, for your use, Supplement No. 3 that deals with the past issues and a number of current issues. Additionally, we have provided for your use, both the Subcommittee and the full Committee, the results of our evaluation of those issues that are applicable to Pebble Springs in light of what has been discussed on the standard Babcock and Wilcox 205 plant, as well as reflecting issues that have come up on the current application using this plant.

I'm pleased to say that we were able to complete our review, provide these to the applicant, and give him an opportunity to state his position with regard to these fourteen issues. Those have been made available to you. Some of these are tailored to fit the Pebble Springs application. However, much of what's contained in there is reflecting those items of discussion that have transpired under the standard BSSAR 205.

Let me go back to the past issues and bring you up to date as to how we stand on the issues that did exist at the time of the last ACRS meeting.

The outstanding issues at that time, as identified under 1.8, Section 1.8, were ten issues. Seven of these issues have been resolved, as identified in Section 1.8 of Supplement No. 3. In addition, we needed to add two more items in this matter in order that we would identify for your purposes what we call outstanding issues. These are (1) turbine missiles, and the fuel handling accident. I'll discuss this brie[illegible].

With respect to Item 1 on containment monitoring system to meet the single failure criteria: at that time we noted that one monitor existed, and that in the event of failure of this monitor there would be no indication in the containment of high radiation. The Electrical Branch at that point felt the need for another monitor.

Later on in the review proces of other plants it was recognized there was a need to examine the fuel handling accident inside containment. We have now provided you in Supplement No. 3 an analysis of the fuel handling accident inside containment. And, in fact, Item 1, dealing with the containment monitoring system to meet the single failure criteria, is also associated with this item.

Now in this Supplement No. 3 we have identified fuel handling accident analysis. In addition to that we have been discussing with the applicant proposed changes to the purge system, including the additional monitoring.

I am now pleased to state that based on our discussions I am confident that we will be able to report that

Items 1 and 5 are resolved, and we will be able to reflect this in the next supplement to the Safety Evaluation Report.

They have, in discussions, met all our concerns, as noted in Supplement No. 3, as well as discussions that we have had with the applicant on response times, mixing analysis, and so forth.

Let me address the item of turbine missiles, of interest to the Committee at the time of our review of February 1976. We had not identified at that time this item as an outstanding issue; the reason being that they identified the fact that if they could not demonstrate that through analytical means there was adequate protection from missiles they would place around the turbine a shield, a shield that would prevent any missiles from damaging safety related equipment.

With that commitment, we proceededthen on the basis that the alternative which was reviewed and accepted would be an alternative to re-orienting the turbine.

You will note in the SER, as well as the supplement, that the applicantdid not wish to re-orient his turbine for reasons of his own, I believe which are primarily economic. He proceeded on that basis. During that period, then, from February until now the applicant completed his analysis and concluded, in effect, that no additional protection was necessary.

The Staff, however, proceeded on an independent basis, made its own analysis, and determined that further protection was required. And let me cite, as a matter of summary, the results in the Supplement No. 3.

We concluded there was a need -- and I shall cite this from the report itself: that the main steamline and feedwater penetration area is the principal contributor to the overall risk from the potential turbine missiles, and that protection of this area from potential turbine missiles would reduce the overall risk substantially if added protection is made in this area

In addition, we stated we felt more protection was needed in the steam generator enclosures within the containment.

Last, we identified the need there for additional protection that would involve, if you will, a steam valve testing procedures program to increase the frequency of testing to provide an added amount of protection from turbine missiles. This matter has been discussed with the applicant, and I think I am at least able to indicate to you a partial resolution of this matter.

First and foremost, the area of the steamline feedwater area which exits from the containment and then faces the turbines, an area in which a large missile could puncture the area in which both Train A and Train B, namely the steam lines are divided into two sections and protected

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by a barrier. If a missile penetrates one of the areas we felt that both areas would be compromised due to the affect of spallation.

After some discussion withthe applicant I can indicate now that they will commit to the installation of a spalling shield inside the walls, first the front wall facing the turbines, and then the side wall protecting Train A from Train B. In the judgment of the staff this appears to be an acceptable solution.

With regard to turbine missiles penetrating containment and into the steam generator area, we have not agreed on a solution. The Staff believes, in its analysis, and in recent discussions with the staff, that a large missile could penetrate this area and could damage the steam generator -- one or more steam generators.

We suggested the possibility of some grill abo[illegible] the steam generators in order to add this protection. At this point in time I believe the applicant would like to talk [illegible] this matter. But first of all let me go through other items. They have rejected our suggestion on the basis of it being impractical.

I note, then, at this point, this matter will have to be further discussed. And, for the present, we will have to continue to identify this item as an outstanding issue.

I failed to mention one item that was not addressed

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in yesterday's discussion of volcanism. In February of 1976 this loomed as a very large issue, an important issue, primarily because of the lack of data on which to draw for information which would give us a reasonable conservative basis for the designing of the plant. Nevertheless, the applicant proceeded to launch into an intensive effort in this area, and I can conclude that we have been able to agree on a conservative basis for the design of volcanic ash at the Pebble Springs site.

All of this information is reported in Supplement No. 3. And for the sake of time I'll skip what this contains, unless there are further questions on this.

DR. BUSH: Are there questions?

(No response)

I guess not.

MR. STAHLE: Let me add one factor not identified on the agenda, but which has been discussed: the matter of security.

We have been discussing this matter with the applicant, and we have pointed out to him the need to update previous work it has done in this area, and have cited the ACRS's interest in this matter. And they have read the various classified reports. However it did not appear appropriate at this time to discuss the area, other than to cite the fact that this matter has been discussed. They are aware of our wbf

requirements, and are pursuing this matter. And I believe we will be able to report to you at the next full Committee meeting.

Other items identified in the agenda: Fire protection. This is under way. The applicant did submit a report. The staff has reviewed it, and provided a series of questions related to this report, and the applicant has told me in November we should have a response to this matter.

I've concluded that this matter is moving along satisfactorily, and I do not foresee any problems as far as Pebble Springs is concerned at this moment.

The item identified as ATWS: Mr. Varga has agreed briefly to touch upon this matter. He's more familiar with the subject matter, and I think you will be interested in the current status at this point in time.

MR. VARGA: As the Committee knows, I think, some two months ago or so, probably longer, Dr. Hanauer gave a brief resume to the Committee about his task force that had been assembled to review all of the past pronouncements and work that the staff had done and to codify it and put it into essentially a position paper that would then go out to the ACRS and then, following that, to the Commission, and then following that, of course, be promulgated as a requirement.

And as the Committee is also well aware, I've been working on this for many years, I think, ever since

WASH-1270. It was in a time period like 1973, I believe.

Last week there was a meeting with Dr. Hanauer and all the principal staff of the NRC, particularly the NRR, the Nuclear Reactor Regulation office. And a report was distributed, a draft-draft report as characterized by Dr. Hanauer, some rather massive and overwhelming looking volumes which contained the summary -- contained the details as well as the summary work that the Staff has under way and the Task Force had under way in coming to grips with putting together all the pieces of the problem.

This is now under principal staff review. There is a schedule that has been established, including two discussions -- several discussions with the ACRS, I believe.

I feel that this matter has taken a significant step forward. I don't bel'eve it is yet ready for public discussion. But I think in the late winter time scale, it's my perception, there are going to be discussions with the ACRS on the details of our recommendation.

DR. BUSH: We'll pick this up as a generic item . In any event it is nice to know that there is progress being made.

MR. ST LD: Let me go on, then, just to at least cite the two remaining items as identified in the Safety Evaluation Report, Supplement No. 3.

With regard to the evaluation of financial qualifications of the applicant: as we cited, this will be done at a later date.

The one remaining item on this list, Supplement No. 3, is also discussed in the handout that was provided to you by Portland General Electric. This is the design of the decay heat removal system. This is a matter that has not -- was not resolved in February of 1976. It continues to be a matter not resolved.

However let me turn your attention to the responses that Portland General Electric has provided as a result of our identifying issues from the, from basically the standard plant of Babcock and Wilcox. These are fourteen iss[illegible], the first of which does indeed identify the decay heat removal system isolation valves. We stated our concern, and you'll see in that Portland General Electric's response. They have simply at this point identified it -- and I think you may wish to discuss thiswith them, the fact that they will meet our criteria. However they are still in the review of this matter and are looking at alternatives means for a resolution of this item.

I must then regard this item as still an open issue.

With regard to the remaining items, although they have been in some cases tailored to fit the Portland General Electric Pebble Springs application, most of these are basic to the responses that the Committee has seen and reviewed with regard to the standard plant of Babcock and Wilcox. However, if you wish, the handout provides a detailed response of the applicant. The Staff has not had a full opportunity to go through these items, and therefore I cannot state unequivocally that we have agreed to all of them. Some are still in review by the Staff as regards to the standard plant.

Fourteen positions, however, are issues that we had informed the applicant are ones that must be responded to. The first twelve require resolution prior to the issuance of the CP. The remaining two items identified in this list would come after that point in time.

DR. BUSH: For the record, you say a handout. Is this an amendment, or is this a pre-amendment handout?

MR. STAHLE: This, sir, in view of the time frame and to meet our goal here of providing the Committee all of the information current, it's a pre-amendment, if you wish. All of it will be documented in an amendment to the PSAR. All of this information and, of course, our evaluation, will be reflected in the next SER. My goal in Supplement No. 4 will be -- will involve all these items as well as the resolution of the 1872 earthquake for Pebble Springs.

That completes my brief summation.

DR. BUSH: Paraphrasing what you've said, then, I am led to conclude that Supplement 4 should, in essence, pick up the seismic aspects. And I guess you still need a letter from USGS on this.

MR. STAHLE: Yes, sir, that's correct.

DR. BUSH: It will also, in essence, pick up those items that I guess interface with BSSAR-205.

MR. STAHLE: That's correct, sir. Fourteen items.

DR. BUSH: And we will then be faced with two or three items that are still under review that have already been cited today; is that the case?

MR. STAHLE: Yes, sir. Of course I have set forth a very ambitious schedule for myself, the Staff, and the Applicant, recognizing that with the schedule for January for the full Committee all this must be completed and submitted on or about the first week in December.

DR. BUSH: But we will be faced with a situation with a relatively unresolved issues: is that right?

MR. STAHLE: Yes, sir, we are down to a few. And I think with the applicant working at the few remaining, it's quite possible that we will be able to cite resolution of all these items.

DR. BUSH: If I were to guess now I would guess the one that has major implications from the design point of view is probably the turbine missile one.

MR. STAHLE: I believe the applicant would like to discuss this, if you have the time. They have provided us brb1

with some new information on their analysis. You may find this of interest. The Staff has not had the opportunity to discuss it; I hope in the next week or two we will, and possibly find a solution to this, what I would consider one remaining item on turbine missiles.

DR. BUSH: Let me, at this stage, ask this questio of the Staff:

I would assume that, providing you can meet, or approximately meet, your goals timewise, then this particular Subcommittee meeting would be such that it would not require another one. In other words, we could remand to the full Committee, utilizing Supplement 4 to the SER, and it would not be necessary to have another meeting to serve as an arbitrator, so to speak.

Is that right?

MR. STAHLE: That was my goal, sir, that I would be able to present to you all the issues, the positions, of the Applicant in our analysis so that we could eliminate the need for an additional Subcommittee meeting.

DR. BUSH: I was proceeding on this assumption.

MR. STAHLE: I have just been reminded I have cited a very ambitious goal of getting Supplement No. 4; but I believe with respect to the nonseismic issues I can get the resources of the Staff in a higher priority. This is quite possible. I have little control of the 1877, sir, at rb3

MR. BROEHL: Yes.

We have cleared three people to review the Sandia report, and they have reviewed it. This was done principally in conjunction with the Trojan plant operating license which we're currently involved in the development of the modified amended security plan for Trojan, and as such we're quite well acquainted with the requirements.

DR. BUSH: It has become "them" now; I believe there are at least two classified reports, if Dr. Lawroski is correct. I simply mention that in passing because this is an item of passing interest to Dr. Lawroski and I'm sure he will ask you about not only one report but others.

MR. VARGA: Particularly Dr. Thompson's report, I believe.

DR. BUSH: Yes. And he asks questions. I only mention this because you do have a couple of months in there, and if you haven't read it he will strongly advise that you do so.

MR. STAHLE: And I have made this point very clear to the Applicant, and I think we are proceeding along these lines, sir.

DR. BUSH: Does the Staff have anything further to add?

MR. VARGA: To be completely objective [illegible] it, just yesterday we had internal discussions. The [illegible] report that I mentioned to you, that is undergoing Staff review, is impacted -- has the potential for impact in many areas. At the present moment it's under review by the Director of Nuclear Reactor Regulation in terms of assessing priorities.

One of the several plants included in that list for potential impact, aside from the 1872 earthquake, is Pebble Springs. The review of the issues that Carl has mentioned, whether this will in fact occur is not yet clear, but there is that potential.

DR. BUSH: Well, that's going to be a long list, Steve, as you and I recognize.

Do either of the other members of the Subcommittee have questions of the Staff at this time?

DR. PLESSET: Could you just in a general way indicate: is is something that seriously affects the design, say, of a new plant, or like Pebble Springs?

MR. VARGA: What, the consideration we have still under --

DR. PLESSET: Yes.

MR. VARGA: No.

My recommendation would be, after looking at the list and considerable discussion internally: I believe in the majority of these cases it's merely documentation. I think the Applicant understands and is well aware, and I think his response indicates that, particularly his handout rb5

that we characterize as a pre-amendment.

The rest of them, including the turbine missile, is his original commitment, which as I understand he still stands by, of providing appropriate protection. He, however, is asking the opportunity to convince us with some additional analyses about the extent of his commitment in terms of pr tection.

Our recommendation, I believe, will be ---even at the present moment -- to proceed to the full Committee with the feeling that this is a plant that we would recommend proceeding for licensing. There are documentation problems that we still have to clear up.

DR. BUSH: I think Dr. Plesset's question was probably specific to ADWAS and its design implications.

MR. VARGA: Oh, yes.

Pebble Springs is no different than the others, and my perception of what the report may conclude won't be anything startling, other than what we've already been concerned about. And there are all sorts of fine tuning -- I certainly haven't read the report in detail, but nothing has startled me from the discussion that I've seen.

DR. BUSH: Thank you. Steve.

Harold, do you have anything?

DR. ETHERINGTON: No.

DR. BUSH: If not, then I believe we are now up to b6

Item 4, the Applicant's response to the NRC Staff report.

I presume that this will tend to expand upon your written handout. Is that the case?

Mr. Broehl, I believe you're the spokesman.

MR. BROEHL: Yes.

We can repond to amplification of the written report to whatever degree you would desire. I would like, in addition, to mention the status of the turbine missile questions. You're aware of the differences in our positions.

DR. BUSH: I think we're prepared to listen to the turbine ones. Let me find out if the Subcommittee has an interest in pursuing specifically an oral discussion of those other issues. I believe we were given this yesterday.

Milt, do you have any items you wish to pursue specifically?

DR. PLESSET: No.

DR. BUSH: Harold?

DR. ETHERINGTON: No.

DR. BUSH: Since it will become a part of the record -- or it is a part of the record of this meeting, the Reporter should have received a copy to attach, and of course it will appear as an Amendment. We can pick those up there.

But we are, I guess, prepared to listen to what you have on the turbine missile.

I suppose I have a vested interest. Maybe I have a conflict of interest and shouldn't listen.

Incidentally, just to set the stage, there have been two turbine failures in the three months.

MR. BROEHL: Unlike the geologists, I have only one exhibit here.

(Laughter.)

DR. BUSH: You're going to confuse us with facts, is that your idea?

MR. BROEHL: Yes.

(Slide.)

You can see in, I believe it is page 3-8 of the SER, Amendment 3, the Staff came up with an estimate of probabilities for a realistic application of our criteria. The Staff requested several modifications in order to make our application acceptable. Those modifications included a small shield, which took care of a small problem in the steam line penetration area; we have committed to this.

In addition, we're looking for a commitment from the manufacturers, a valve testing program. This we've agreed to.

Further, we have committed to an inspection program to improve the integrity of the turbine discs.

In addition to that, we have, of course, diverse bS

redundant turbine trips.

Now, these additions really affect the F-1 probability of the overall reduction.

Another item that they had in there was that we could look to analyses to further reduce the very low numbers that are in there. The only two numbers that we seemed to be concerned about were related to the control room and the penetration of the containment.

The control room is down; I don't recall the exact number. It's on the on the order of 1.6×10^{-7}. They came up for the estimate of strike that would cause damage in the steam generator compartment of 2.7×10^{-7}. We reanalyzed using that criterion and the MIS code and came up with 2.17×10^{-7}, which is in very close agreement with the Staff's conclusion.

One thing that was not modeled by the Staff w[illegible] the steam generators. The Pebble Springs arrangement between the time of the original application went in and the present situation, we went from two to four moisture separator reheaters.

Now these reheaters happen to be located adjacent to the LP stages so they in effect provide additional shielding between the turbine and the containment. And when you apply the ballistics criteria through the moisture separator reheaters, it reduces that number down to 1.5[illegible] $\times 10^{-7}$, which is very close to the measured number, 10^{-7}.

Those are the numbers allowing no credit for the improvements we have made in the basic turbine design and ea2

the testing of the valves.

Now I don't know of anybody who has really put a number on these. The best that I've been able to come up with is Dr. Matt Taylor of NRR who did the turbine missile probabilities work for the WASH-1400 study. And Matt felt that the benefits you could get on this were in the order of two-ish or three-ish, to use his words, just for the valve testing program.

That number applied to his, of course, brings it below the 10^{-7}.

There are a number of other conservatisms in there, and I think just a judgment type is we are close enough to the 10^{-7} that I think anything to reduce it still further is just, you know, unless it were a fairly reasonable cost and something that would not interfere with plant operation, we just don't believe it's realistic.

We think we have a good number and we ought to be able to build a plant and operate it as designed.

DR. BUSH: Let me raise a point that is peripheral to this, more so than anything else, just to, I guess, indicate some of the other variables that might arise.

These are B&W plants. They differ from other PWRs in their once-through steam generators which have positive aspects and they may have negative aspects.

In at least two instances with Babcock and Wilcox

eb3

designs, there has been cracking, selective cracking, of turbine disks, probably due to plastic concentration at a given position in the low-pressure stage. The significance of this I don't think has really been established.

However, it does represent a potential failure mechanism and it could exist not only in the overspeed stage but could also occur at normal operating speeds.

I do not intend to express a judgment. I guess I'm only asking a question, if you're aware of this, and if measures are being considered in the long run to eliminate or mitigate this particular problem.

MR. BROEHL: Yes, we are aware of the potential problem, and we have two activities which we look forward to, to minimize any potential. One is certainly maintaining an extremely high quality in the water chemistry, and the other is through in-service inspection of the turbine disks.

DR. BUSH: Of course the problem with stress corrosion is because of its insidious nature, the amount of credit you can get for in-service inspect isn't quite so high as you might like to have because it may happen very rapidly.

I believe with a plant that is some years down the stream it becomes an academic issue. I'm am only raising it at this time. Obviously you're aware of the problem, and presumably by the time this plant comes on line, there may be a definitive solution to the problem, but I think it has ep4

the potential of being a serious problem.

That's the only issue I raise.

Do you have rebuttal or something?

MR. CHRISTENSEN: I just wanted to mention that we are following this very closely with regard to the company and in doing so, instead of pumping drains forward through our feedwater system, we are designing it to pump backwards. So we are not concentrating maybe in the area of 20 times, which seems to be the historical concentration in the low-pressure turbine, and going through the moisture separator.

So we are aware of it and we are minimizing this effect.

DR. BUSH: I simply wished to raise the issue. I think it's recognized and I hope action will be taken. But one has to recognize that this is a potential mechanism, and it must be pursued appropriately.

I won't express an opinion one way or the other on where you are with regard to the ragged edge of your probabilities here, your accumulated probabilities. I think if this new study of failure probability picks up a factor of two to four, which it may do, if we can reduce probability, that makes your situation look a little better.

Presumably these are cumulative probabilities for both high trajectory and low trajectory missiles, so I

eb5

suspect you are probably controlled more by the low trajectory missile in this situation.

MR. BROEHL: Yes.

DR. BUSH: Harold, do you have anything you wish to add?

Do you have any further comments in this area?

MR. BROEHL: No.

DR. BUSH: Does PGE wish to introduce anything else in the record orally?

(No response.)

DR. BUSH: Well, I think we understand the situation with regard to the non-seismic issues. As I understand it, there is either resolution or convergence toward resolution on plant-specific items. And with regard to generic items, at least there are positions on some of the critical ones. Obviously there can't be on all of them.

With regard to the items-- I guess this isn't really a retrofit, is it, or a ratchet with regard to the BESAR-205? That has very specific connotations.

I would say with regard to the seismic, this has been pretty thoroughly aired, and I can't speak for what the Committee will do. One obvious solution would be to write a generic seismic letter relevant to the Northwest but a priori, I can't indicate what will happen. That seems to be the most logical approach.

eb6

I guess if there are no further items we can conclude an interesting two days.

(Whereupon, at 1:50 p.m., the meeting of the ACRS Subcommittee on Pebble Springs was concluded.)

MISSILE STRIKES
ON STEAM GENERATORS

NRC ESTIMATE 2.7×10^{-7}/year

MIS CODE:

WITH NRC ASSUMPTIONS $2.17 \times 10^{-7} \pm 0.3 \times 10^{-7}$

WITH NRC ASSUMPTIONS PLUS MODELING OF MOISTURE-SEPARATOR-REHEATERS (MSRs)

$1.56 \pm 0.26 \times 10^{-7}$

MISSILE STRIKES
ON STEAM GENERATORS

NRC ESTIMATE 2.7×10^{-7}/year

MIS CODE:

WITH NRC ASSUMPTIONS $2.17 \times 10^{-7} \pm 0.3 \times 10^{-7}$

WITH NRC ASSUMPTIONS PLUS MODELING OF MOISTURE-SEPARATOR-REHEATERS (MSRs)

$1.56 \pm 0.26 \times 10^{-7}$

Oct 27, 1977

PGE RESPONSES TO NRC OPEN ITEMS
(TO BE RESOLVED PRIOR TO A CP)

OPEN ITEM NO. 1

Decay Heat Removal System Isolation Valves - This issue is identified in Section 7.4.1 of the Pebble Springs SER. The power system arrangement to the DHR system suction isolation valves is such that during a pipe break outside containment, one train of the system cannot be isolated from the Reactor Coolant System assuming the single failure of one electrical bus. An acceptable design would be separate Class 1E power supplies for each valve as was recently submitted in B-SAR-205.

Response to Open Item No. 1

The Decay Heat Removal System (DHRS) will be designed so that the system can be isolated from the Reactor Coolant System (RCS) in the event of a pipe break outside Containment assuming the single failure of one electrical bus.

PGE recognizes an acceptable design to be separate Class 1E power supplies for each DHRS suction isolation valve. However, PGE intends to evaluate alternative designs, fully meeting the NRC isolation criterion above, which may be more compatible with current plant design.

We plan to submit a proposed scheme for incorporating the NRC isolation criterion into the DHRS design in a future PSAR amendment.

OPEN ITEM NO. 2

Overpressure Protection at Low Operating Temperatures - The staff is currently developing a position which will provide requirements for the design of a protection system for these events. The applicant must commit to the following minimum criteria prior to issuance of a CP:

(1) Credit for operator action. No credit can be taken for operator action until 10 minutes after the operator is made aware that a transient is in progress.

(2) Single failure criteria. The pressure protection system should be designed to protect the reactor vessel, given any event initiating a pressure transient, and followed by a single active component failure. Redundant or diverse pressure protection systems will be considered as meeting the single failure criteria.

(3) Testability. Provisions for periodic testing of the overpressure protection system(s) and components shall be provided. The program of tests, and frequency or schedule thereof, will be selected to assure functional capability when required.

(4) Seismic design and Standard 279-1971 criteria. Ideally, the pressure protection system(s) should meet both Seismic Category I and Standard 279-1971 criteria. The basic objective, however, is that the system(s) should not be vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

(5) Reliability. The system(s) provided must not reduce the reliability of the emergency core cooling system or residual heat removal systems.

Response to Open Item No. 2

Appendix G, 10 CRF 50 specifies the minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB. These requirements are provided in the Technical Specifications for normal and test conditions as a pressure limit which varies as a function of reactor coolant temperature and rate of temperature change.

Overpressure protection against exceeding these limits is provided by (1) the pressurizer safety valves and (2) the decay heat removal system relief valves.

As shown in Figure 1, the pressure limit is relatively low at low reactor coolant temperatures and then increases to higher limits at higher reactor coolant temperatures. The Appendix G limit shown in the figure was determined using the following:

(a) The methods outlined in topical report BAW-10046, "Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G".

(b) RT shift predictions from Regulatory Guide 1.99.

(c) Actual base metal and weld wire material data (typical).

(d) Predicted accumulated radiation for 32 effective full-power years.

(e) Design heatup and cooldown rates.

This is a typical limit curve for B&W plants and is applicable to Pebble Springs.

The pressure limit is bounded by the pressure setpoints of the pressurizer safety valves and DHRS suction line relief valves so that when the reactor coolant pressure limit is above the setpoint of the pressurizer safety valves, overpressure protection will be provided by the pressurizer safety valves; and when the RCS pressure limit is below the setpoint of the pressurizer safety valves, overpressure protection will be provided by the DHRS suction line relief valves. This is ensured by requiring, in the Technical Specifications, that all four DHRS suction valves located between the RCS and DHRS be normally open during DHRS operation when the pressure limit is below the setpoint of the pressurizer safety valves. In addition, the temperature range at which overpressure protection is required by the DHRS suction relief valves coincides with the temperature range for operation of the DHRS.

The Appendix G limits are restrictive only at the lower reactor plant temperatures associated with RCS startup and shutdown operations. Worst-case pressure transients that could occur under these conditions were determined to be the following:

(a) RCS makeup valve stuck full open.

(b) Three HPI pumps actuate.

(c) All pressurizer heaters energize.

(d) Loss of cooling water to DHRS coolers.

(e) Core flood system valve opens.

(f) Start of a reactor coolant pump with hot water in secondary side of the steam generators.

These transients have been analyzed, and the results are given in the following sections (the majority of the information is already contained in Pebble Springs PSAR Section 9.3.5.4.1.4). The results show that the pressurizer safety valves and DHRS relief valves provide mitigation for all transients assuming (1) a single active component failure, (2) no operator action, and (3) loss of offsite power.

With the DHRS isolation valves open and the system operating, the previously listed transients or incidents can increase the RCS pressure and thus the pressure in the DHRS. These incidents have been analyzed to determine the maximum required relief capacity to prevent the DHR pump suction pressure from exceeding 500 psig. The basis for analyzing the various transients was as follows:

(a) DHRS is placed in operation during plant cooldown at 305°F RCS temperature.

(b) Pressurizer water level is at the normal level for power operation.

(c) Pressurizer pressure is at the midpoint of the allowable band for starting the DHRS.

(d) No credit is taken for any spray into the pressurizer or any steam relief from the pressurizer.

(e) No credit is taken for letdown flow rate to the Makeup and Purification System (MPS).

Following is a brief description of each incident analyzed:

(a) Loss of DHRS Cooling - It is assumed that DHRS cooling is lost by a loss of power to the DHR pumps, loss of cooling water flow to the DHR coolers, or two independent single failures causing inadvertent closure of one suction valve in each DHRS train.

The rate of RCS pressure increase for the postulated latter incident is relatively slow, allowing ample time for an operator to take action. An analysis of the transient showed that the operator has far more than the required 20 minutes to assume corrective action.

Two conditions of initial level and pressure in the pressurizer were examined, which would correspond to expected conditions during cooldown for RCS temperatures below 170°F using the pressurizer model of the CADD code. Conditions for reactor coolant temperatures higher than 170°F were not considered because of the large pressure margin between the DHRS operation and the Appendix G limit shown in Figure 1. The analysis assumed, conservatively, that the reactor coolant expands in a completely isolated system with the maximum potential decay heat of 1 percent of full power with only partial credit taken for energy absorption by the RCS metal. The analytical results are as follows:

Case 1 - Low Pressure (initial conditions representative of conditions that exist on cooldown at 50 psia)

Time, minutes	Pressurizer pressure, psia	Pressurizer level, ft
0	50	31.3
10	53.5	32.8
20	56.3	34.4

Case 2 - High Temperature (initial conditions representative of conditions that exist on cooldown at 170°F)

Time, minutes	Pressurizer pressure, psia	Pressurizer level, ft
0	100	25.1
10	105.8	26.7
20	109.7	28.3

This transient would be indicated by any of the following:

i. Increasing reactor coolant pressure (meter indication).

ii. Increasing reactor coolant temperature (meter indication).

iii. Low flow in DHRS (alarm and meter indication).

iv. DHRS pump cavitation.

v. Closed valve indication for the DHRS letdown valves (valve position indication).

The operator can mitigate the transient by any of the following methods:

i. Reopen one of the DHRS isolation valves so that relief valve protection is restored.

ii. Open pressurizer electric-operated relief valve.

iii. Increase pressurizer spray flow.

iv. Open manual pressurizer vent to drain tank.

v. Open letdown valve of MPS.

vi. Initiate auxiliary feedwater operation.

(b) Makeup Control Valve Fails Full Open - It is assumed that the makeup control valve controlled by the pressurizer level controller malfunctions and goes full open. For this analysis, no credit was taken for letdown to the MPS. The high flow rate through the makeup valve into the RCS plus the seal injection flow rate increases pressurizer water level thus increasing pressure.

(c) All Pressurizer Heaters Energized - It is assumed that, although pressure is increasing, all pressurizer heaters are energized and remain on. For this analysis, a minimum pressurizer water level is assumed as it would produce the fastest pressure rise and highest relief rate requirement. To stop the pressure increase, a minimum outflow rate from the pressurizer must be created such that the heater capacity can generate an amount of steam equal to the additional vapor space being created by the outflow rate. This required outflow rate is 911 gpm. The decreasing pressurizer water level will cause the makeup valve to go full open (with pressurizer level controller in "automatic"); this inflow rate (see above, Makeup Control Valve Fails Full Open) must be added to the 911 gpm to obtain the relief requirement. When the pressurizer water level decreases to the low-level heater cut-out setpoint, the heaters will be automatically deenergized.

(d) High Pressure Injection System Accidentally Actuated - It is assumed that the entire HPIS is actuated. For the analysis, it was assumed that all three HPI pumps pump into the RCS through the HPI lines, increasing pressurizer water level and thus increasing pressure.

Two pumps operate in one HPI string and one pump in the other. The three HPI pumps are powered by two separate electrical buses. Normally the design is such that only one pump can operate on a single bus which results in a maximum of two HPI pumps operating simultaneously. For conservatism and safety, the DHRS relief valve is sized for all three HPI pumps operating.

(e) Core Flood Tank Outlet Valve Accidently Opened - It is assumed that the core flood tank (CFT) outlet valve (motor-operated) is accidentally opened. The CFT outlet valve was closed during plant cooldown before RCS pressure approached 600 psig. When it is accidentally opened, CFT water will be pushed into the RCS increasing pressurizer water level and thus pressurizer pressure until CFT pressure and pressurizer pressure equalize. Even with pressurizer pressure initially at the high end of the band for starting the DHRS, the resultant pressurizer pressure is not high enough to (1) exceed the DHRS design pressure by more than 10 percent, or (2) exceed the limits of 10 CFR 50, Appendix G, even with the decay heat pump operating at shutoff head. For this incident, relief by the decay heat letdown line relief valve is not required. The relief valve could be actuated depending upon initial pressurizer pressure, because pressure at the decay heat pump suction could reach the relief valve setpoint. For this transient, the reactor coolant pressure increases rapidly to the equilibrium pressures listed in Table 1, assuming no credit for relief valve action.

(f) Starting a Reactor Coolant Pump - The initial conditions of this transient result from the filling of the steam generator with 470°F feedwater with the rector coolant at refueling temperature. This condition is reached during steam generator filling operations assuming that the feedwater controls fail and the operator continues to fill the steam generator with feedwater in excess of the allowable 225°F feedwater temperature. The temperature of the feedwater in the steam generator reaches 240°F, as does the primary water at elevations above the lower steam generator tubesheet. This is a result of heat transfer from the secondary to the primary side of the steam generator during filling. It is assumed that the primary water below the lower tubesheet remains at refueling temperature. At the end of the filling operations, a reactor coolant pump is started. The peak expansion rate is approximately 1750 gpm, which is less than the DHRS relief valve capacity.

As the analysis results presented in Table 1 show, the required minimum capacity for DHRS letdown relief valve is 2000 gpm. At a setpoint of 455 psig, minimum required capacity is 2000 gpm at 10 percent accumulation. This relief valve will prevent the DHRS design pressure from being exceeded by more than 10 percent during the worst incident concurrent with the DHR pump operating at any developed head up to and including shutoff head. Each of the dual DHR letdown lines contains a relief valve sized for the full relief flow rate in the event that the DHRS is being operated on one letdown line only. Normally, both letdown lines are used and the available relief capacity is therefore two times greater than the requirement.

Steam condensation in the pressurizer during the insurge was not used in the analysis. Actually, steam will condense during the compression of the steam bubble; accounting for the steam condensed would result in a lower rate of pressure rise than shown above. However, it would not affect the calculated required relief capacity listed in the table. The relief valve must relieve a volume rate equal to or greater than the insurge volume rate that flows into the pressurizer at the maximum desired pressure. The required relief capacities listed in the table have been determined in this manner.

TABLE 1

PRESSURE RISE AND RELIEF VALVE CAPACITY

Incident Description	Rate of Pressure Rise - psi/min	Required Relief Capacity - gpm
Loss of DHRS Cooling	<10 psi in 20 minutes	225
Makeup Control Valve Fails Full Open	36	524
All Pressurizer Heaters Energized	5	1,425
HPIS Accidentally Opened (all three HPI pumps operate)	162	2,000

	Equilibrium Pressure - psig	
	At Pressurizer	At DHR Pump Set
CFT Outlet Valve Accidentally Opened:		
Initial pressurizer pressure at midpoint of band	432	455
Initial pressurizer pressure at high point of band	474	497

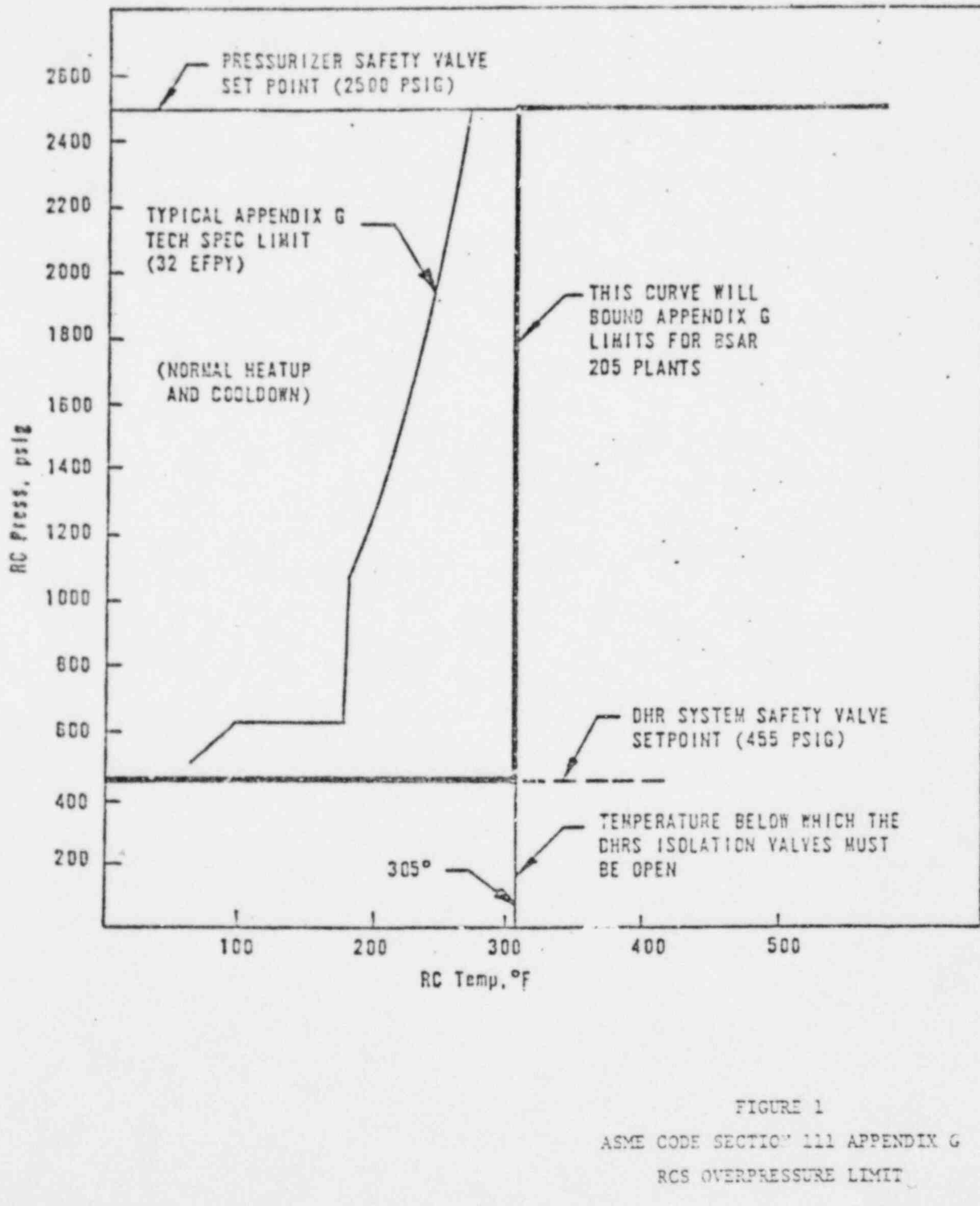

FIGURE 1

ASME CODE SECTION III APPENDIX G

RCS OVERPRESSURE LIMIT

OPEN ITEM NO. 3

ECCS Analyses - The applicant references topical reports BAW-10102, BAW-10074, and BAW-10104 for the analyses in accordance with 10 CFR 50.46. Since the approval, by the staff, of the emergency core cooling system model described in BAW-10104, several changes to the model have been submitted by Babcock & Wilcox and approved by us. To have a referenceable worst-break which is wholly in conformance with 10 CFR 50.46, Appendix K, and to ensure that the selected postulated breaks adequately define the "worst-case" situations, additional analyses are required. (These analyses were requested in a Commission letter from S. Varga of NRC to James H. Taylor of B&W, dated May 10, 1977.)

Response to Open Item No. 3

The worst-case large break analysis requested in the letter from S. Varga to J. H. Taylor of May 10, 1977 was submitted by letter from J. H. Taylor to S. Varga on September 30, 1977. The results of this LOCA limits analysis show a peak cladding temperature at the core midplane (6 ft. elevation) of 2114°F, which is 12°F less than calculated previously and reported in BAW-10102, Rev. 2 (Table 7-1). The letter from S. Varga of May 10, 1977 also requested one (worst-case) small break analysis. This analysis is scheduled to be submitted to the NRC by the end of 19[illegible]. Based on a comparison of the results of the large and small break spectrum analyses presented in Pebble Springs PSAR Tables 15.13-17 and 15.13-18, the peak cladding temperature for the worst-case small break should prove to be significantly lower than calculated for the worst-case large break.

The analyses performed and reported in BAW-10102 and the letter of September 30, 1977 assume a higher flow rate and a higher power level than that reported in the Pebble Springs PSAR, as explained in Section 15.13.[illegible] An analysis has been performed for a plant similar to Pebbl[illegible] [illegible]rings using B&W's August 1977 ECCS evaluation model. Three parameters were changed (power level, RCS flow, and

Containment pressure) from those reported in the letter from J. H. Taylor of September 30, 1977. The power level and RCS flow were adjusted to correspond to 102 percent of the rated power level of 3600 MWt and nominal RCS flow of 103,500 gpm, the same values as used in the Pebble Springs PSAR. The Containment pressure used in the analysis showed a peak of 28.5 psig. The peak Containment pressure appropriate for the Pebble Springs ECCS analysis is 30.8 psig (Figure 3.4-2 of BAW-10102, Rev. 2), which indicates that the analysis is conservative for Pebble Springs. The results of this LOCA limits analysis yielded a peak cladding temperature of 2059°F at the core midplane. This is 67°F below the equivalent temperature reported in BAW-10102, Rev. 2, and 55°F below the temperature reported in the September 30, 1977 letter. All other criteria of 10 CFR 50.46 were also shown to be satisfied.

OPEN ITEM NO. 4

<u>High Pressure Injection Line Break</u> - A break in a high pressure injection line (HPI) between the reactor coolant system piping and the last HPI check valve results in a small LOCA. We require additional information to evaluate the consequences and necessary operator actions to mitigate the consequences of this event.

<u>Response to Open Item No. 4</u>

The Pebble Springs makeup and purification/HPI pumps suction and discharge header design has been revised (see letter dated September 7, 1977 from W. J. Lindblad, PGE, to S. A. Varga, NRC). The Pebble Springs HPI System, as modified, is presently functionally identical to the corresponding design described in B-SAR-205. Consequently, the discussion in Section 6.3.2.17.2 of B-SAR-205, which describes the necessary operator action required to mitigate the consequences of a postulated break in the HPI line between the RCS piping and last check valve, is applicable to Pebble Springs.

Information will be provided in a future PSAR amendment to demonstrate that the Pebble Springs and B-SAR-205 HPI System designs are equivalent insofar as the HPI injection line break analysis is concerned.

OPEN ITEM NO. 5

Provisions for Shutdown - The applicant must demonstrate that the plant can remain for a prolonged period in a hot shutdown condition assuming loss of off-site power and using only safety-grade equipment or show that the plant can be cooled and depressurized using only safety-grade equipment (assuming loss of off-site power) to the level required for decay heat removal system actuation.

Response to Open Item No. 5

PGE will provide information in a future PSAR amendment to demonstrate that the plant can remain for a prolonged period in a hot shutdown condition assuming loss of off-site power and using only safety-grade equipment or that the plant can be cooled and depressurized using only safety-grade equipment (assuming loss of off-site power) to the level required for DHRS actuation.

OPEN ITEM NO. 6

Makeup Line Break - The applicant must evaluate the required actions and consequences resulting from a break in the normally pressurized makeup line considering all potential single active component failures.

Response to Open Item No. 6

The Pebble Springs makeup and purification/HPI pumps suction ar discharge header design has been revised (see letter dated S 1977 from W. J. Lindblad, PGE, to S. A. Varga, NRC). The HPI System, as modified, is presently functionally ider corresponding design described in B-SAR-205. Consis'

B-SAR-205 design, the third makeup pump and supporting BOP auxiliaries in the Pebble Springs design can be manually transferred to either ESF safety train and will be normally aligned with the same train as the standby HPI pump. Consequently, the response in B-SAR-205 to NRC Questions 212.147 and 212.227, which describes the required operator actions and consequences resulting from breaks in the normally pressurized makeup line and other locations in the HPI pump discharge header, assuming a concurrent single active failure, is applicable to Pebble Springs.

Information will be provided in a future PSAR amendment to demonstrate that the Pebble Springs and B-SAR-205 HPI System designs are equivalent insofar as the makeup line break analysis is concerned.

OPEN ITEM NO. 7

Passive Failures - It is our position that detection and alarms be provided to alert the operator to passive ECCS failures during long-term cooling following a LOCA which allows sufficient time to identify and isolate the faulted ECCS line. The applicant will be required to commit to the staff's position (see also the detailed discussion in NUREG 0138).

Response to Open Item No. 7

Pebble Springs design provides assurance that a postulated passive failure during the post-LOCA recirculation phase will not degrade ECCS capability or contribute significantly to post-LOCA accident doses.

Potential sources of passive leakage include pump and valve seals and instrument fittings. Leakage from the sources noted would be expected to be less than 1 gpm. However, for conservatism, it is assumed that a 50-gpm leak rate is possible. Excluding the Containment penetration area, potential sources of leakage are localized in the Containment spray pump and DHR pump and heat exchanger compartments. Leakage in these compartments will be routed to the Seismic Category I, stainless steel-lined Auxiliary Building A or B sumps. High-level Seismic Category I sump indications are provided in the control room. Therefore, high sump level would indicate a passive failure in either safety train A or B, allowing appropriate operator action to isolate the affected train. This isolation procedure would not degrade ECCS capability as the redundant DHR and Containment spray trains would be more than adequate to satisfy ECCS requirements.

In addition, an ESF filtration system is provided that takes suction on the Containment spray pump and DHR pump and heat exchanger compartments. This system is automatically aligned on an ESFAS signal and is designed in accordance with Regulatory Guide 1.52. Therefore, substantial passive recirculation mode leakage will not contribute significantly to post-LOCA doses.

The amount of inventory loss from the Containment due to the passive leakage does not constitute an appreciable hazard to ESF pump NPSH requirements. Even at a 50-gpm leakage rate, Containment recirculation level would decrease at a rate less than 1 in./hr. Furthermore, low-level Seismic Category I Containment sump indication is provided in the Control Room to alert the operator to any significant sump level decreases.

OPEN ITEM NO 8

Essential Manual Valves in ECCS - Experience has shown that consideration must be given to the possibility that, prior to an accident, locally manual valves (handwheel) might be left in the wrong position and remain undetected. The staff will require remote position indication in the control room for all such manual ECCS valves, the mispositioning of which could compromise ECCS performance.

Response to Open Item No. 8

The status of essential manual valves in the ECCS will be verified in the following manner:

(a) Those valves which have to be operated during the sequence of a normal plant start-up and/or shutdown will be provided with position switches and will be monitored as part of the "inoperable status" indication for that system.

(b) Those valves located in normally accessible areas will be visually verified for correct position at least once every 31 days in accordance with administrative procedures.

(c) Those valves located in normally inaccessible areas, where routine visual verification is undesirable, will be provided with position switches and will be monitored for correct position from the control room.

The above criteria are consistent with Regulatory Guide 1.47.

OPEN ITEM NO. 9

Excessive Heat Removal Events of Moderate Frequency - The applicant is required to show that no fuel damage occurs for such events (DNBR >1.32). Therefore, it is not appropriate to reference the main steam line break analysis which shows DNBR <1.32 at 3.1 seconds into the transient.

Response to Open Item No. 9

Excessive heat removal from the RCS can result from a maloperation or inadvertent operator adjustment of the feedwater control system which causes a reduction in feedwater temperature or an excessive increase in the feedwater flow. These transients have been analyzed in Section 15.10 of the PSAR and shown to result in a DNBR >1.32.

An excessive heat removal accident could also result from the inadvertent opening of a steam safety, atmospheric dump, or turbine bypass valve by the operator or an equipment malfunction such as a pressure regulator failure. The steam pressure regulator malfunction or failure resulting in increasing steam flow has been analyzed in Section 15.1.36 of B-SAR-205. For no fuel damage to occur, these malfunctions or failures are limited to a maximum 15-percent step increase in steam load. The increase in steam flow resulting from a stuck open main-steam safety or modulating atmospheric dump valve would be limited to about a 6-percent and 7-percent step increase, respectively. However, a maloperation of the turbine bypass system may result in a steam flow greater than 15-percent rated. To ensure that the rated steam flow is limited to a maximum 15-percent step increase, interlocks (or alternative means) will be provided to prevent incidents of moderate frequency from causing spurious opening of the atmospheric and condenser dump valves. The interlocks will not inhibit valve operation for a turbine trip, generator trip, or load rejection when dump valve operation is desirable.

OPEN ITEM NO. 10

Decay Heat Removal System Cooler Bypass Valves - The rate of cooldown is normally controlled with these valves. The concern is that loss of air to these valves, causing them to fail closed, may result in maximum flow being directed through the coolers. This could result in an excessive cooldown rate of the reactor coolant system.

Response to Open Item No. 10

The DHRS cooler bypass valves V044 and V045 shown in PSAR Figure 9.3-9, are pneumatic operated valves which fail close on loss of air. To reduce the probability that a loss of motive power could result in an excessive cooldown rate of the RCS during operation of the DHRS, the pneumatic operators will be modified to incorporate a "fail in position" feature. The details of this revision will be reported in the FSAR.

OPEN ITEM NO. 11

Feedwater Isolation - The main feed system contains two headers, one for each steam generator. Each header contains only one safety-grade feedwater isolation valve which receive redundant ESFAS signals. The applicant must show that the failure of this valve is considered in those Chapter 15 events requiring feedwater isolation.

Response to Open Item No. 11

The Chaper 15 accident analyses have included consideration of the failure of one of the FWIV's to close on demand. For those accidents where feedwater isolation is required (the most limiting being a main-steam line break), the feedwater control valves, which receive redundant buffered ESFAS closing signals, provide a suitable backup to the FWIV's. Additional protection is provided by buffered ESFAS signals that trip the turbine-driven feedwater pumps. Credit may properly be taken for these nonseismic Category I backup devices since the consequences of accidents involving spontaneous secondary piping failures are significantly lower than those involving primary piping failures. This position is in accordance with the NRC policy set forth in NUREG-0138.

OPEN ITEM NO. 12

Chapter 15 Events - The applicant will be required to provide a discussion for each Chapter 15 event describing all of the actions required in the recovery mode following a transient. Our interest is in evaluating the operator's role in achieving and maintaining stable conditions. An example of such a situation would be the necessity of the operator to secure the HPI pumps after a steam line break to prevent repressurization of the reactor coolant system at low temperatures.

Response to Open Item No. 12

PGE will provide a discussion for each Chapter 15 event describing major actions required in the recovery mode following a transient such that the operator's role in achieving and maintaining stable plant conditions can be evaluated. This information will be provided in a future PSAR amendment prior to issuance of the construction permit.

PGE RESPONSES TO NRC OPEN ITEMS (TO BE RESOLVED AFTER CP)

OPEN ITEM NO. 1

Credit for Nonsafety Grade Systems - The applicant must show that no credit is assumed for nonsafety-grade systems for the mitigation of any Chapter 15 event. For example, the turbine trip analysis in the PSAR assumes power runback by the ICS. Table 15.0-3 shows the "Equipment Assumed Functioning in the Accident Analysis".

Response to Open Item No. 1

General - The use of nonsafety-grade equipment for anticipated transients is shown in Table 15.0-3 of the PSAR to be limited to the use of the (1) turbine trip (via CRDCS) and/or (2) the turbine bypass system. The anticipated transients (of moderate frequency) so identified in the table are (a) rod group withdrawal at startup and at power, (b) control rod misoperation, (c) CVCS malfunction (boron dilution), (d) turbine trip, (e) loss of normal feedwater, (f) excessive heat removal, (g) inadvertent operation of the ECCS, (h) loss of four pumps, (i) break in primary system penetration lines, and (j) control room uninhabitability.

For the above-listed transients, except turbine trip, a reactor trip is initiated prior to tripping the turbine. The turbine trip transient is the only one in the category of a turbine trip prior to reactor trip. Analyses of the turbine trip transient with and without turbine bypass or automatic runback of the reactor have been performed for the B-SAR-205 application which demonstrated an acceptable secondary system pressure response (see Section 15.1.7.2.4 of B-SAR-205). This analysis is also applicable to Pebble Springs.

The transients where the reactor trip occurs prior to turbine trip and subsequent use of the bypass system are discussed below.

Use of Turbine Bypass System - For transients where a reactor trip is initiated prior to tripping the turbine, the heat demand for approximately 2 seconds following turbine trip is the same whether or not the bypass system is assumed to function. In addition, the heat demand used in the Pebble Springs analysis more closely approximates the heat demand without turbine bypass than the heat demand with turbine bypass. In view of the above two facts, it is not surprising that recent analyses performed with the heat demand simulating no turbine bypass system action showed negligible differences to the analysis presented in Section 15 of the Pebble Springs PSAR. This statement can be summarized as follows:

> "Turbine bypass used as part of its normal role but is not a required function. Adequate secondary steam pressure relief capacity is available without bypass action through the atmospheric dump and/or safety valves with negligible effect on transient response (except for turbine trip analysis where failure of the bypass is discussed)."

Use of Turbine Trip via CRDCS - The anticipated transients identified as Item h, loss of four pumps and Item i, break in primary system penetration lines, cause a reactor trip prior to turbine trip. The conclusions reached in the discussion above, which show negligible difference in results, remain valid.

Furthermore, since these two transients are characterized as undercooling transients, assuming that a turbine trip occurs is conservative since less heat will be removed from the primary system, accentuating the undercooling. Item i, control room uninhabitability, is dependent on the initiating event and is therefore covered by the spectrum of postulated events identified in Table 15.0-3.

The other undercooling events identified above that assume turbine trip with bypass system action are (a) rod group withdrawal at startup and at power, (b) control rod misoperation, (c) CVCS malfunction, and (e) loss of normal feedwater. The assumption of turbine trip would be conservative for these undercooling transients. Item g, inadvertent operation of the ECCS transient, although not an undercooling transient, is an overpressure event and again the assumption of turbine trip is conservative since removing less primary heat results in higher primary pressures.

The remaining transient, (f), excessive heat removal, is an overcooling transient which causes a decrease in primary system pressure. It is not as obvious for this transient what effect turbine trip has on the results. The safety limit of concern for overcooling transients is the DNBR and the necessity of not exceeding the limit of 1.32 (BAW-2) for transients of moderate frequency. To demostrate that turbine trip does not have a significant effect on the DNBR, an analysis with a 205-FA plant, 3672-MWt initial condition was completed. The transient analyzed was a steam pressure regulator malfunction. Two cases were run: one assumed turbine trip and the second assumed no turbine trip. The reactor power, pressure, and DNBR are plotted over the time frame in which minimum DNBR occurs in Figure 212.239-1 of B-SAR-205. The results show that in consideration of MDNBR limits the effects of turbine trip versus no turbine trip are negligible for overcooling transients.

The above information will be incorporated into Chapter 15 of the FSAR as appropriate.

OPEN ITEM NO. 2

Boron Dilution Events - The applicant must provide additional analyses of the boron dilution events considering the plant conditions other than power operation or refueling (as specified in Standard Review Plan 15.4.6). In addition, they must discuss all potential dilution sources.

Response to Open Item No. 2

Dilution water is supplied to the RCS by the MPS in both the Pebble Springs and B-SAR-205 designs. These systems have identical interlocks and alarms to prevent improper operation as described in Section 7.7 of the Pebble Springs PSAR and B-SAR-205. Alarms are provided to annunciate that the interlock setpoints have been reached.

The current Pebble Springs MPS design does not incorporate a makeup tank bypass line as reflected in the B-SAR-205 MPS design. However, the B-SAR-205 analysis, which considers plant conditions other than power operation or refueling, should bracket Pebble Springs since greater dilution rates are assumed and the potential dilution sources are equivalent.

PGE will provide additional analyses in the FSAR of boron dilution events considering plant conditions other than power operation or refueling (as specified in Standard Review Plan 15.4.6).