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DEC 30 1983

MEMORANDUM FOR: Frank Miraglia, Assistant Director for Safety Assessment
Division of Licensing

FROM: R. Wayne Houston, Assistant Director for Reactor Safety
Division of Systems Integration

SUBJECT: WAPWR REVIEW

Reference:
1. R. W. Houston, "WAPWR Review," NRC Memorandum to F. Miraglia, October 18, 1983.
2. "WAPWR Preliminary Reference Standard Plant, Primary Side Safeguards Module," Westinghouse Nuclear Energy Systems (no report number or date, provided to NRC during meeting of June 13, 1983)

Plant Name:	Westinghouse Advanced Pressurized Water Reactor (WAPWR)
Docket Number:	None
Licensing Stage:	Pre-PDA
TAC No.:	668
Responsible Branch:	SSPB
Project Manager:	G. Meyer
DSI Branch Involved:	RSB (Primary Responsibility)

In reference 1, we provided a brief summary of the review of reference 2. This memo provides additional information regarding reference 2 and documents our September 26 and 27 meeting with Westinghouse. Our questions provided the agenda for the meeting. The answers are our perception of the Westinghouse responses.

We are now performing a detailed review of Westinghouse's module and will provide formal questions on it. While no formal action by W on this memo is required, we suggest you transmit it to them for their information.

We also point out that, based on our questions and perception of the W Responses, the initial module submitted had an extraordinary amount of errors, wrong figures, mistakes, typos, etc.. Reviewing documents of this poor quality is unnecessarily time consuming on our part and indicates a potential

CONTACT: W. Lyon, X29405

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lack of good QA on submitted documents. I strongly suggest you bring this to Westinghouse's attention so that they are aware of it and can try to correct the problem on future module submittals.

Original Signed By
R. Wayne Houston

R. Wayne Houston, Assistant Director
for Reactor Safety
Division of Systems Integration

Enclosure

cc:	R. Mattson	David K. Moran, SSPB
	G. Meyer	Nick Fioravante, ASB
	Chief, AEB, DSI	Joseph P. Joyce, ICSB
	Chief, ASB, DSI	T. Dunning, ICSB
	Chief, CSB, DSI	C. G. Tinkler, CSB
	Chief, ICSB, DSI	Glenn Kelly, CSB
	Chief, CPB, DSI	D. R. Lasher, RRAB
	Chief, RRAB, DST	Jacques Read, AEB
	Chief, MEB, DE	Jocelyn Mitchell, AEB
	Chief, SGEB, DE	H. L. Brammer, MEB
	Chief, GIB, DST	S. P. Chan, SGEB
	Chief, PSRB, DHFS	T. M. Su, GIB
	Cecil O. Thoms, DL	F. Liederbach, PSRB

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SURNAME				RWHouston			
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DSI Branch Involved:	RSB (Primary Responsibility)

In reference 1, we provided a brief summary of the review of reference 2. This memo provides additional information regarding reference 2 and documents our September 26 and 27 meeting with Westinghouse. Our questions provided the agenda for the meeting. The answers are our perception of the Westinghouse responses.

We are now performing a detailed review of their final module and will provide formal questions. No DL action is needed regarding this memo since the information will be covered in the detailed review.

R. Wayne Houston, Assistant Director
for Reactor Safety
Division of Systems Integration

Enclosure

cc: See Next Page

12/22 12/27/83

OFFICE	CONTACT: W. Lyon	DSI:RSB	DSI:RSB	DSI:RSB	DSI:AD:RS
SURNAME	X29405	WLyon:jf	JRosenthal	BSheron	RHouston
DATE		12/15/83	12/15/83	12/23/83	12/ /83

ENCLOSURE: CP. 15 QUESTIONS
AND RESPONSES

1. (page 15.0-1) We note W states the module covers all of the accidents that are significantly impacted by the ISS. We further note W addresses only accidents of the design basis type. These are inconsistent observations. We intend to examine the WAPWR with respect to accidents beyond the usual design basis. We also are not convinced that all of the design basis events have been covered, although the preliminary examination shows that most are.

   A. See answers which follow.

2. (15.1-1) What is the basis for identification of the events that are limiting cases?

   A. The sensitivity studies have not been accomplished, and what is presented is based upon prior work with other plants. This probably will be an open item, and it may be moved to the secondary side module. The stuck rod assumption is also under evaluation, and may be "held" to provide investigation time.

3. (15.1-3) Operation of the rod cluster control assembly banks is stated to be restricted in such a way that addition of positive reactivity in a secondary system steam release accident will not lead to a more adverse condition than the case analysed. What is the restriction means, and what is the logic supporting the conclusion?

   A. This refers to an operational restriction.

4. (15.1-3) The negative moderator temperature coefficient used for the analysis is stated to be presented in Figure 15.1-1, but this figure is labeled "later." Hence, we have no data upon which to base an appraisal.

A. Part of the approach in preparation of the document was to obtain an acceptance type review from NRC, hence there are areas where the submittal was not complete. W will provide the information.

5. (15.1-3) The LOFTRAN code is stated to be applied to this analysis. Since this is a new design with changes in a number of components, we require additional information pertinent to the application of the code. This is to include all differences in nodalization, and significant changes pertinent to elevations, volumes, and heat transfer areas. We anticipate a need to perform audit calculations in the future which, at a minimum, will involve selected thermal-hydraulic NSSS responses (not necessarily of the LOFTRAN type, although this is not excluded).

   A. W pointed out there is a LOFTRAN SER that was just prepared by RSB. (The intent of a portion of the comment is to identify something that the staff will expect in the future.)

6. (15.1-4) Please explain how assumption of a hot shutdown condition is sufficent to establish that the design DNBR will not be exceeded.

   A. The statements are based upon plant experience and upon the content of reports that have not been completed and submitted.

7. (Reference C, p. 6) Why is the boron concentration not the same in all of the system components?

   A. The Cp. 6 information should be 2500 ppm. This will be corrected.

8. (15.1-4, 15.1-5) "This is the maximum...." "The assumed steam release is typical...." Which?

   A. Maximum should be used in both locations.

9. (15.1-5) "Boron solution at 2500 ppm enters the RCS providing sufficient negative reactivity to prevent core damage." Do you mean to imply this is the only concern with regard to damaging the core?

A. No. There are many other concerns.

10. (15.1-5) Figures 15.1-2 through 15.1-4 provide no useful information since they are labeled "later." Hence, we cannot complete our review since the information is not included in the submittal.

    A. These are to be completed. See 4.

11. There is no discussion of decrease in feedwater temperature (15.1.1), increase in steam flow (15.1.2), and excessive increase in steam flow (15.1.3). The inadvertent opening of a single steam generator relief valve or safety valve with failure to close (15.1.4) is stated to be most limiting. This conclusion is presented without justification. Since this is a basic objective of the review, as outlined in the SRP, justification of the conclusion should be provided.

    A. To be provided. See 4. This information will be provided in Module 6 which covers the secondary side.

12. (15.1-6) Is there any significance to the wording in item 1 of 15.1.5.1 that "the core remains in place and intact" as compared to the wording in the regulations (such as DC35) which places emphasis upon maintainability of coolability?

    A. W stated there is no significance in the difference in wording. Wording consistency will be provided.

13. (15.1-8) In item 1 at the bottom of the page, what is the implication if less xenon exists than would correspond to the equilibrium condition?

    A. The situation is for zero power, and therefore there is no xenon.

14. (15.1-9) W, in Section 1.3.2.3, stated they intend to eliminate the stuck rod assumption, because taking all of the conservative assumptions "is unreasonable.... Westinghouse plants ...have never experienced the failure

of a rod to drop into the core on a demand signal.... (and) it can be shown that the probability of getting a stuck rod is very low." We do not accept this argument on the basis of the presented material, and hence do not accept that GDC 26 has been met. We also would inquire whether there was a demand signal in the Salem failure to scram. Now, on p. 15.1-9, W is discussing behavior near the stuck rod. Which is it--stuck or not?

A. If W elects to use the lack of a stuck rod as an approach, additional backup information will be provided. We forsee some difficulty in this approach, and in fact it may require W having to request an exception to, as a minimum, GDC 26 and 27. A portion of the difficulty with this material (contained in the W submittal) is that it was boiler plate type information applicable to other W plants, and was provided for completeness, although not necessarily applicable to the WAPWR. This will be corrected.

15. (15.1-9) "This core analysis considered no moderator feedback from power redistribution and nonuniform core inlet temperature effects." What are these effects?

A. This is a typo. All are considered. It will be corrected.

16. (15.1-10) Does W consider the modeling of the SI system as described in WCAP-7907 to be appropriate for vessel injection in WAPWR?

A. The reference has sufficient flexibility that it applies to vessel injection.

17. (15.1-10) Please provide additional information regarding the adequacy of the item 4 assumption (conservative?) for the event being discussed.

A. W stated that for practical purposes the assumption means that the steam generator tubes are always covered, that the steam exiting the steam generators is dry, and showing that during a transient the actual overall heat transfer coefficient (U) is less than used here makes the fouling factor usage effectively conservative.

18. (15.1-10) Do all of these realistically happen at once and is the assumption fully conservative?

    A. The feeling at W is that there are so many conservatisms that one doesn't need the stuck rod assumption. With respect to the stuck rod, the present thought is to keep the reactivity, and they are thinking about the effect on peaking. The entire item is undecided.

19. (15.1-11) What is the complete meaning of the first sentence (i.e., what are the conditions)?

    A. The parameters or references will be provided.

20. (15.1-11) The staff has not accepted the stated assumptions to all be conservative, and hence does not accept the Results conclusion.

    A. The lack of a stuck rod may not be assumed. The future report will show the conservatism.

21. (15.1-11) The core heat flux is not shown as stated.

    A. This will be provided. See 4.

22. (15.1-13) Please explain why the assumption of RCPs running is more conservative than RCPs tripped with respect to not exceeding the design DNBR.

    A. This Q was withdrawn. The answer is provided on the prior page of the W document.

23. (15.1-13) What is the referenced iodine partitioning? See also Q 34.

    A. Refer to Table 15.1-2, III C, II C.

24. (15.1-14) Where are the tech spec iodine concentration values documented that you reference?

A. The middle column of Table 15A5 of Appendix 15 is the Tech Spec value. The table numbering is incorrect and will be corrected, as will the technical specification information.

25. (15.1-15) What site was used for the referenced calculations? Were these done with WAPWR or for some other plant?

A. Vogtle (Georgia Power) was used for the site information since it is the most recent available, and is in accord with the SRP. Information pertinent to LOCA and SGTR was adjusted for the WAPWR, but the SLB was not. (The steam dump into the EWST was not used.) References will be provided. The steam release is for the standard 412 plant.

26. (15.1-15) How was the 0.12% defective fuel determined?

A. ANS-N237 uses 0.12% for the source term. The Tech Spec is 1 μCi/gm et al. (ref table). They will provide feedback on damaged fuel.

27. (15.1-16) What is the substantiation for the conclusions regarding filters? Where is this work reported?

A. This is boiler plate information. The design basis LOCA was used. Information will be provided later, probably as part of the balance of plant submittals.

28. (15.1-20) What is the implication if a SGTR condition results rather than the assumed 1 gpm leakage?

A. The maximum primary to secondary leakage for four generators, according to Tech Spec, is for 0.35gpm from one steam generator. A simultaneous SLB with SGTR may be in the range of CFR dose limits. This work will be redone for the secondary side submittal, although that specific accident may not be.

29. (Fig. 15.1-5) What is the cause of the discontinuities in the RCS pressure curve?

    A. The pressurizer empties at 1750 psi, and the SI is initiated at 1000 psi. The accumulator comes on at 600 psi, and the steam generator empties at 150 sec. Note also the core boron curve does not correspond to the pressure curve. These are typical curves and do not necessarily represent WAPWR. Actual information for WAPWR will be provided.

30. We are not clear as to whether the reactor returns to (or becomes) critical during the overcooling accidents. Please clarify the conditions.

    A. This depends on the stuck rod assumption. If a stuck rod is assumed, then it will become critical.

31. Is the assumption made that charging pumps are available and in use for the injection of boron during this and the LOCA investigations? Would they be used in WAPWR under these conditions?

    A. Charging is not used for LOCA for analysis purposes. During a LOCA, the charging header is isolated, but delivery to the RCP seals continues provided off-site power is available. If off site power is lost, the positive displacement (PD) pump comes on to provide seal protection. (Note the PD pump is also capable of use in the event of a total loss of AC power for the WAPWR.) An additional aspect of the PD pump with WAPWR is that CCW is not required for pump operation. The PD pump is cooled by the fluid it is pumping.

32. Is the elimination of the BIT of any significance with respect to overcooling accidents?

    A. The BIT tank helps in that it keeps the core sub-critical in the event of the credible steam line break. The WAPWR will return to criticality.

33. You indicated that for the selected conditions, no DNB occurs. Are there any steam line breaks for which DNB does occur?

    A. No. The selected conditions are the most conservative.

34. W states "The results presented are a conservative indication of the events which would occur assuming a secondary system steam release...." How is this true with respect to releases since the assumption was made that perfect moisture separation occurs in the steam generator, which prevents the carryout of contamination contained in the liquid phase (additionally assuming that leakage from the RCS to the SG secondary is occurring)?

    (Not discussed)

35. (15.6-1) What is the pressurizer level situation if the selected orifices are not for normal letdown?

    A. Pressurizer level normally would be in the correct range, and if not, the makeup would be operating. See the response to Q37. The maximum letdown flow rate for WAPWR is about 250 GPM, and two makeup pumps would have to be in operation to keep up.

36. (15.6-1) Do you consider it impossible to have a flow path via SI piping toward the BWST?

    A. The design of the BWST has changed, and the tanks have been eliminated in the WAPWR design. Effectively, the storage is now inside containment as a part of the EWST. The RHR line is the only one that W would think could be a path.

37. (15.6-1) Can a break occur downstream of the letdown flow meter such that a low flow alarm would not occur?

    A. Yes. Note the makeup would be running to keep up and there would be a radiation alarm in the auxiliary building. The latter would lead to an isolation action.

38. (15.6-2) If letdown were in use prior to the break, would the operator be concerned regarding frequency of makeup system operation?

A. Yes. Makeup running too often is of concern.

39. (15.6-2) Turning to the referenced Table 11.1.1-2, we find the words "No portion of this chapter (11) is pertinent to the WAPWR Primary Side Safeguards System."

(not discussed)

40. (15.6-2) With regard to "...complete severance is somewhat conservative.", what is the real difference between that and a longitudinal split? As compared to several tubes ruptured due to a common cause failure?

A. This question was deleted. Clearly, it is not conservative, but W is looking at the design basis here. I left it up to them to do whatever they want with reference to the question.

41. (15.6-3) What is the sensitivity of steam flow/feedwater flow mismatch in regard to detection of SGTR?

A. Four to seven percent of the feed flow could be associated with one tube. This translates to about a 1/4 inch deviation on the readout. The sequence of events will be changed in the W report.

42. (15.6-3) What is an overtemperature N-16 signal?

A. This comes from a combination of N-16 power, cold leg temperature, and pressure. The proper wording is "low DNBR signal."

43. (15.6-4) Item e is not clear in that no quantitative information is provided to back up the 60 minutes statement. We further cannot evaluate the paragraph unless secondary side descriptive information is provided. Is the passive steam condenser to be used? If so, it would appear to be very beneficial during SGTR.

A. This topic is to be moved to the secondary side module so that the necessary information will be available. I plan to insist that the stated 60 minutes be shown to be realistic. W probably will have an open item here, and will probably reference an upcoming analysis document. Probably, the single failure approach will be used here, and the multiple failure covered in the PRA. W will cover the 60 minutes in some manner. (I indicated no problem with reference to future work and an open item.) Note they have decided to elminate the passive condenser from the secondary side.

44. (15.6-4) In regard to Item f, we note there are many dependencies in addition to the one mentioned. Is there a reason W mentioned this particular one?

A. W will rewrite the statement.

45. (15.6-4) In item c, are you planning to use PORVs as the preferred depressurization technique? If so, what is the intended handling of the SI and charging systems? Control of pressurizer level?

A. Pressurizer spray is the preferred mode. Use of the PORV is a safety grade backup. The pressurizer is larger in the WAPWR as compared to prior plants, and the intent is to avoid opening of the PORVs during transients if this can be practically achieved. Aux spray capability will be provided in WAPWR.

46. (15.6-5) Please expand on your statement that the WAPWR is designed to prevent the steam generator from filling up with water under these circumstances (See also the first part of Q 43.)

A. For the case of the single tube SGTR, level would remain on scale for the example, based on present information. (This may not be the final answer.)

47. (15.6-5) Please provide a comparison of real and conservative evaluation for the listed parameters in the last paragraph.

A. Only conservative information is provided and planned for the FSAR. This would be done for an environmental analysis. It is also a customer option.

48. (15.6-8) How is it conservative to assume 0.35 gpm goes to the faulted steam generator?

A. See 28.

49. (15.6-8) Will WAPWR be considered operable if condenser vacuum pump discharge monitors are inoperable? (See item c on P. 15.6-3.) Will WAPWR have steam line monitors? Note item D may not be conservative with respect to recognition time if the operator depends upon blowdown monitors.

A. Availability of steam line monitors was assumed for some of the RT analyses. The actual system that will be provided is open and not selected. They are not depending upon blowdown monitors. The page 15.6-3 item c working is not really applicable, and again represents the use of boiler plate in the early submittal. This will be corrected.

50. (15.6-8) In item g, what site and what conditions are being referenced?

A. The calculations are according to the method of Reg. Guide 1.145. See 25.

51. (15.6-9) In the first paragraph, we do not agree this has been estabished since we were unable to find the referenced paragraph.

A. Previously answered.

52. (15.6-9) We are unable to find the referenced Table 15.6.3-4, and therefore have no basis for review of the last paragraph.

A. Should be Table 15.6.3-2; to be corrected.

53. (15.6-10) What is the tech spec leakage limit for unknown leakage location? Known?

A. Identified leakage is 10 GPM, and unidentified is 1 GPM (not steam generator tube leakage).

54. (15.6-10) We are unable to find the referenced Table 15A-7.

A. Should be 15A-6, not -7. To be corrected.

55. (15.6-10) Note the reference to Table 11.1-2 which you state in Chp. 11 not to be applicable.

A. Another reflection of the early use of boiler plate to reflect on the type of information to be provided. This information will be provided in A15.

56. (15.6-10) Please explain item i which identifies an 8 hour leak duration in light of several references in the text to 60 minutes.

A. The steam generator with the SGTR was assumed to be isolated in 60 minutes. The intact steam generators were assumed to steam for an assumed 8 hours.

57. (15.6-11) Why is it necessary to release so much steam to the atmosphere from the ruptured tube steam generator? We cannot fully understand and evaluate unless we know the secondary configuration.

A. The tabulated values are incorrect with respect to steam release. An additional assumption was made here, namely that the passive steam generator cooling did not work (an additional failure). Note that the passive steam generator system would be applicable if it were provided, but this has been removed from the WAPWR.

58. What is the time of reactor trip and the parameter which caused the trip?

A. Radiation monitors have not been determined (and this calculation may change?).

59. What release occurs via blowdown?

A. W has assumed that all of the noble gases are released with steam and therefore escape. The remainder is caught in the blowdown system water processing.

60. (Fig. 15.6.3-1) We are unable to find the referenced Table 7.5-1.

A. Either the material will be provided or the reference will be deleted.

61. (Fig. 15.6.3-1) This figure shows steam generator level as being used for identification of the faulty steam generator. Please discuss sensitivity. The text lists a number of parameters the operator would use for identification but we do not recall level. Why?

A. The figure will be changed or the text expanded. The level rise rate is one to three percent per minute during the time one is on the safety valves or the SG PORVs.

62. (Fig. 15.6.3-1) This shows operator control of SI on pressurizer level. Please explain.

A. W will also show subcooling and/or cover in the discussion.

63. (Fig. 15.6.3-2) Please explain how the leak is terminated with this primary pressure history. Note also Fig. 15.6.3-5.

A. The figure will be replaced. This is used for the upper bound on dose, and should not be presented at this location. Note the leak cannot be terminated if the information presented in the figure is correct.

64. (Fig. 15.6.3-6) This figure appears to show a negative flow at time zero. Please explain. Is this figure consistent with Table 15.6.3-1, item III.a.?

A. This figure is with the passive SG cooling system. Negative steam flow is a code fluke early on. The water level reaches the steam separators at 644 sec. Note the level may be off scale on the instrumentation at the end of one hour.

65. (Fig. 15.6.3-8) Why is feedwater flow continued to the faulted steam generator? (See also Figs. 15.6.3-11 and 12.)

A. This is in error and the figure will be removed. The figure should be the feed plus the leak, which in reality is the leak. This also does not show an RCS depressurization, and hence is misleading. A more meaningful example will be provided.

66. (Fig. 15.6.3-9) Is this "real world" behavior for safety valve flow rate?

A. No. One would expect open/close behavior. Correct behavior to be provided.

67. (Fig. 15.6.3-10) Please note this behavior in light of Q 63 as contrasted to the text discussion.

A. This is inconsistent and will replaced.

68. (15.6-14) In light of the wording in item c, would it be correct to state "The localized cladding oxidation limits of 17 percent are not exceeded."? Why do you not reference thickness?

A. This is the statement used for all W SARSs. The wording will be changed to be consistent with the rule.

69. (15.6-14) In item b., why is the reference not to hydrogen generation as stated in the rule?

A. See Q68.

70. (15.6-15) Why is the core temperature referenced as reduced rather than being maintained at an acceptably low value as stated in the regulations?

A. See Q68.

71. (15.6-17) What is the justification for the statement that the EWST level will always remain above the minimum required level? Suppose there is a significant leak outside of containment.

A. W takes all voids in containment into account for the small break, with the RCS full, with no allowance for water contained in the tanks (accumulators and core flood tanks), and with water flowing back into the EWST. They also allow for one pump house to be full. There is still enough water in the EWST so that the level is sufficient to meet NPSH requirements. The assumption is made that saturated water exists above the level needed to meet the EWST volume requirements.

72. (15.6-18) What is the length of the downcomer in WAPWR as compared to typical existing W plants?

A. Thirty feet as compared to 16.

73. (15.6-22) Please comment on the applicability of the work in Ref. 15.6.5-19 to the different geometry of the WAPWR.

A. The base cases are based on the assumption that topicals are applicable. W plans to conduct reanalyses and sensitivity studies in 1984 to cover some aspects of this area. It will be an open item in the PDA work.

74. (15.6-22) What is the meaning of "conservatively calculated" as stated with respect to the decay heat?

A. The approved Appendix K ANS plus 20%.

75. (15.6-24) When you state that no fuel clad damage occurs, do you also mean that there is no ballooning or rupture?

A. Code calculations show no ballooning and no rupture. The time at or above 1800 °F is very short.

76. (15.6-25) Please discuss the leakage of recirculating sump solution that you mention in light of the WAPWR geometry that we understood would essentially eliminate this contribution to release.

A. Recirculation leakage is only eliminated if the pump house is a part of the WAPWR. The pump house has a closed cooling and ventilation system. Note that a path from the pump house to the containment is only opened in the event of an RHR system line break. The relief back into containment is at about a 10 to 15 psi differential pressure from the pump house to the containment. The vent area is sized so that the pump house pressure remains at or below 60 psig for the rupture of the largest RHR line (DEB) at 400 psi. The release rates assumed are based on the same considerations as applicable to containment. The liner plate will be eliminated in the concept, which will slightly increase the release rate. (An epoxy coating will be used.) Note also that the pump house is not considered to be a part of the containment. Nevertheless, the pump house provides a significant reduction in the frequency of releases, and a significiant reduction in off site dose. A difficulty in the real operational world during operation of a pump with a leaking seal is that the filters become loaded and ineffective, and therefore operation of the pump must be terminated. The pump house eliminates that problem. Another problem with accident condition operation may be the attempt to close isolation valves with crud in the system. There is no guarantee that they will adequately close and not leak. The pump house concept would allow isolation of the subsystems. Operation could be continued on one while others could be cleaned up and repaired. The pump house was included in the calculation of results.

77. (15.6-15) What is the containment purge that is mentioned as contributing to release at the beginning of the accident? We note this is mentioned on

p. 15.6-28, with a reference to Section 6.2, but we are unable to find the referenced material.

A. The reference to Section 6.2 is incorrect. This normally would be Section 9. It will be corrected.

78. (15.6-28) Are the pump compartments sealed during operation? Are they vented with termination of venting upon establishment of a need and if so, how is the need determined? Please discuss.

A. They are totally isolated. See 76.

79. (15.6-29) Prior questions regarding the site apply.

A. See prior answers.

80. (15.6-32) How long do the CRTs inject water? Do they also inject N?

A. The CRTs inject for 10 to 20 minutes typically, and nitrogen is injected. W will think about the situation with nitrogen injection for conditions of slow depressurization.

81. Section 15.1.5 appears to be very cursory.

A. The PRA will cover much of this. W will change the wording.

82. Section 5.2, decrease in heat removal, is not covered.

(Not discussed)

83. Section 15.5.1 appears to be cursory, what happens with the charging pumps?

A. This section really is not applicable for SI due to the shutoff head. The effect of the charging system was not considered. This will be added.

84. In Section 15.6, do you consider that breaks can also involve opening of valves?

A. Yes. For example, the PORV is considered. Inner and outer isolation valves for the RHR are subjected to periodic leak testing. Automatic actuation of the PORV block valve is being considered, but there would need to be an over-ride for purposes of feed and bleed cooling of the RCS. The RHR break is beyond the design basis, and is considered from a risk point of view. See also Q36. (Note one would like to be able to "push a button" and be in the once through cooling (feed and bleed) mode.) The WAPWR design currently shows three PORVs, but W may change to two. The once through cooling mode is a consideration. There are also two hot leg vents that can be used for feed and bleed operation. Note also the high head SI pumps presently reach zero flow at 1800 psi. W is going to move SGTR and SLB discussion into the secondary side module.

ENCLOSURE 2

Response Breakdown

(1) Requested information provided or question answered: 1, 2, 3, 4, 5, 6, 7, 8, 9, 12, 13, 14, 15, 16, 17, 18, 23, 24, 25, 26, 28, 29, 30, 31, 32, 33, 35, 36, 37, 38, 41, 42, 45, 46, 48, 49, 50, 51, 52, 53, 54, 56, 57, 59, 61, 64, 66, 68, 71, 72, 73, 74, 75, 76, 78, 79, 84

(2) Question probably will result in an open item: 2, 6, 43, 73

(3) Information to be provided in the near future: 4, 6, 10, 11, 19, 20, 21, 25, 26, 27, 28, 29, 43, 55, 58, 60, 61, 80, 81, 83

(4) Reference 2 to be changed or corrected: 7, 8, 12, 14, 15, 24, 41, 44, 49, 52, 54, 60, 61, 62, 63, 65, 66, 67, 68, 69, 70, 77, 81, 34

(5) Question deleted by staff representative: 22, 40

(6) Question not covered: 34, 39, 82

(7) Essentially a staff representative view of future need: 1, 5, 11, 40, 43

(8) Information not provided: 47, 58