OCT 21 1986

MEMORANDUM FOR: Victor Nerses, Project Manager
PWR Project Directorate #5, DPL-A

FROM: Carl Berlinger, Chief
Reactor Systems Branch
Division of PWR Licensing-A

SUBJECT: SEABROOK STATION RISK EVALUATION PERTINENT TO EMERGENCY PLANNING

REFERENCES:

1. "Seabrook Station Risk Management and Emergency Planning Study", Pickard, Lowe and Garrick, Inc., PLG-0432, December 1985.

2. "Seabrook Station Emergency Planning Sensitivity Study", Pickard, Lowe and Garrick, Inc., PLG-0465, April 1986.

Plant Name:	Seabrook Station, Units 1 and 2
Docket Number:	50-443, 50-444
Responsible Branch:	PWR Directorate #5
Project Manager:	Victor Nerses
Review Branch:	Reactor Systems Branch, DPL-A
Review Status:	Ongoing

We have read the reference documents and have discussed their contents with personnel from Brookhaven National Laboratory (BNL), who are assisting the Staff in evaluation of the subject issue. We have agreed with BNL that we will follow certain potential containment bypass accidents while BNL will be responsible for other bypass conditions. This principally involves our taking responsibility for the steam generator as a bypass path resulting from core overheating phenomena, with BNL taking responsibility for bypass paths which potentially initiate core melt accidents, such as LOCA outside containment. We also agreed with BNL that we would pursue certain questions pertaining to the subject issue, and that they would pursue others.

The Enclosure to this memorandum represents our response to the split of work between ourselves and BNL. We suggest this be forwarded to the Applicant with the suggestion that it serve as the basis for conversations between the Applicant, ourselves, and perhaps BNL.

The major concern with steam generator behavior during a core melt accident is the rupture of multiple tubes in response to high Reactor Coolant System (RCS) temperatures which follow core uncovery. This accident sequence is of concern any time there is a core melt with the Reactor Coolant System at more than a few hundred psi pressure, with no water in the SG secondary side. These conditions lead to a potential for natural circulation transport phenomena to

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significantly heat the tubes prior to breach of the reactor vessel. The resulting loss of tube strength can lead to tube rupture. Reactor Coolant Pump operation, as outlined in many plant emergency procedures, almost assures this to be a concern. If tube rupture occurs, and any of the secondary side valves are open, the secondary side is breached outside containment, or the reactor coolant system pressure is above the SG relief valve setpoints, then containment is bypassed.

A number of questions need to be addressed on this issue in order to approach resolution with respect to the Seabrook investigation. Several of these are provided in the Enclosure. We anticipate further questions will arise as the investigation proceeds.

Original signed by:

Carl Berlinger, Chief
Reactor Systems Branch
Division of PWR Licensing - A

Contact: W. Lyon, x27940

Enclosure: As stated

cc: C. Rossi
R. Ballard
V. Benaroya
J. Milhoan
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M. Hodges
S. Long
G. Bagchi
B. Doolittle

RSB:PWR-A	RL RSB:PWR-A	for RSB:PWR-A
WLyon	RLobel	CBerlinger
10/20/86	10/ 20 /86	10/21/86

TECHNICAL EVALUATION OF THE EFZ SENSITIVITY STUDY FOR SEABROOK

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6. Site Consequence Model

* indicates that date was changed or section was reistated by NRC
# indicates NRC wants information to be changed or added.

10/22/86

AGENDA FOR BRIEFING ON REVIEW OF SEABROOK EPZ STUDY

1. BNL Report

    a. schedule

    b. contents

2. Potential Weaknesses in PSNH Study

    a. Event V

        1. check valve failure rates

        2. scrubbing credit for submerged release

        3. potential for becoming dominant contributor

    b. S/G Tube Failure Due to Overheating

        1. company model -> will not occur

        2. research work -> may be dominant risk contributor

3. Other Issues Significant to Review

    a. BNL Assessment of Containment Challenge

    b. PLG study on impact to risk from containment capability corresponding to 1% strain

    c. Dominant Dose Curve Release Catagory (S2W) Isotopic Ratio

4. Caveats

    a. Completeness

    b. Closeness to Criteria

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 23 1986

Docket Nos. 50-443/444

Mr. Robert J. Harrison
President & Chief Executive Officer
Public Service Company of New Hampshire
Post Office Box 330
Manchester, New Hampshire 03105

Dear Mr. Harrison:

Subject: Request for Additional Information for Seabrook, Units 1 and 2, Emergency Planning Sensitivity Study

The enclosed Request for Additional Information (RAI) is supplemental to the RAI dated October 8, 1986, except that question 20 is a restatement of the question 20 posed in the earlier RAI. Several of the questions in this RAI document the oral questions raised in our meetings on October 15 through 17, 1986. Additional questions pertaining to other areas of our review have also been included. Please provide your responses promptly to facilitate our review.

Questions or additional information regarding this matter should be directed to the Technical Project Manager for the review of the Seabrook Emergency Planning Sensitivity Study, S. M. Long (301) 492-8413.

Sincerely,

Steve M. Long

Steven M. Long, Project Manager
PWR Project Directorate No. 5
Division of PWR Licensing-A

Enclosure:
As stated

cc: See next page

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Enclosure

REQUEST FOR ADDITIONAL INFORMATION
SEABROOK STATION, UNITS 1 AND 2
DOCKET NOS. 50-443 AND 50-444
EMERGENCY PLANNING SENSITIVITY STUDY

Restated Question:

20. Assess the impact on risk of assuming that the containment capability corresponds to the pressure which produces 1% strain in the containment wall.

Additional Questions:

29. The S2W release category isotopic distribution listed in the Table 4-3 in PLG-0465 shows release fractions of cesium and tellurium that exceed the release fraction for noble gases and greatly exceed the release fractions for elemental and organic iodine. Please justify the isotopic distribution of the S2W release category consistent with WASH-1400 source term methodology.

30. The S7W release category isotopic distribution reflects a decontamination factor (DF) of 1000, for all isotopes except noble gases, because the release point is submerged in the RHR vault. WASH-1400 source term methodology credited BWR releases with a DF of 100 when they occurred through a subcooled suppression pool, but set the DF to 1 when the pool was at saturation temperature.

    a. Discuss the degree of subcooling that would be expected in the RCS water that pools in the RHR vault following blowdown through the RHR system.

b. Justify the use of a DF=1000 in light of the WASH-1400 methodology and the degree of subcooling expected in the RHR vault water.

31. Provide a typical calculation to demonstrate that small diameter penetration sleeves do not punch through the containment wall under the worst pressure condition assumed in the analysis.

32. In your prediction of large deformation behavior of the containment, full bond was assumed between reinforcement and concrete between two adjacent vertical cracks; assess the effect on containment behavior including penetration capability, if no bond stress is assumed between the reinforcing steel yield point and ultimate strength of steel. Based on our discussions in the meeting, it is our understanding that you will perform this assessment assuming no bond stress.

33. Confirm that a complete and independent check will be performed for the containment strength calculations that served as the basis for the EPZ sensitivity study.

34. Fully address the effect of uncertainty in the ultimate strength of Cadweld splices on the pressure capacity of the containment. As discussed in the meeting, your response should address potential, non-ductile failure of the Cadweld splices.

35. Assess the response of the containment sump encapsulation vessel on the containment integrity.

36. Discuss the results of recent EPRI tests to address the potential for strain concentration in the liner at crack locations.

37. Demonstrate that your calculations fully account for the differences in stress-strain behavior between the reinforcing steel and the ~~lower~~ liner plate with regards to strain compatibility.

38. Quantify the leak areas associated with other containment failure modes as discussed in Section 5 of Appendix H to the PLG report #PLG 0300. Also, assess the impact on risk by assuming these failure modes to be type A

rather than type B failures including the effect of simultaneous occurrence of various failure modes.

39. Only selected penetrations were analyzed in the calculations; compile a list of all containment penetrations, categorize according to behavior and demonstrate that each penetration is adequately covered by the analyses that have been performed.

40. What indications are available if RHR is lost during shutdown (e.g. spurious closure of suction valve)?

41. What indication is available for vessel level during shutdown and refueling modes?

42. Does loss of power to the pressure transmitter that provides input to the autoclosure interlock for RHR suction valve cause the valves to close?

43. To what level(s) is the RCS drained for maintenance activities while shutdown with fuel in the vessel? What level is necessary to maintain connection with the ultimate heat sink?

44. Describe the availability of the SI pumps while shut down. How difficult would it be to restore the SI function to respond to transients during shutdown and refueling conditions? Consider maintenance of the SI system in your response.

45. Provide the procedures for establishing cold overpressure protection when shutting down.

46. Is the primary system made water-solid during shutdown?

47. Address the risk from creep failure of the steam generator (S/G) tubes due to exposure to high temperatures during core melt sequences in which the reactor coolant system (RCS) remains at high pressure and the secondary sides of the S/Gs are dry. Your discussion should reflect the recent experiments and modeling efforts that show 3-dimensional convective flows

which transfer heat from the overheating core to other places within the RCS, particularly into the upper plenum and from the upper plenum along the hot legs into the S/Gs and through the U-tubes. Also include the influence of pressure driven flows resulting from reactor coolant pump (RCP) seal LOCAs, PORV/safety valve actuations, "bumping" the RCPs, etc. Localized heating effects due to redistribution of fission products in the RCS should be included.

a. What is the total probability of occurrence for the high RCS pressure core melt sequence with dry S/Gs?

b. What is the estimated conditional probability that the S/G tubes will fail due to overheating before the pressure is relieved by failure of the RCS elsewhere?

c. What is the effect of preexisting S/G tube leakage (within technical specifications) on the heating rate and temperature required for failure of the leaking tube(s).

d. What release category would creep failures of the S/G tubes result in?

48. Most of the work pertinent to severe accidents has addressed plant behavior at full power, on the assumption that this represents the major contribution to risk. Also, WASH-1400 assumed containment failure was probable following a core melt, making containment bypass sequences relatively less important. Therefore:

a. Please address the possibility of accidents inside the containment building while in Modes 2-6 (Startup, Hot Standby, Hot Shutdown, Cold Shutdown, and Refueling) insofar as these accidents could impact upon risk. In particular, consider the effect of reduced safety equipment availability and containment integrity requirements permitted by technical specifications while shutdown or refueling.

b. Event V and steam generator tube rupture provide a direct path from the RCS to the environment during severe accidents. Please describe the Seabrook work which identifies any other direct paths.

c. Please provide further information and/or specific references pertinent to release of radioactive material located outside of the containment building (e.g. spent fuel pool, radwaste systems) insofar as the magnitudes are large enough to impact upon the issue under consideration here.

49. The FSAR gives RHR relief valve flow rate as 900 gpm with a set pressure of 450 psi. The flow rate does not agree with the value used in Reference 1, section 3, page 6. Please explain.

50. Please describe the mechanism for assuring that plant changes and new knowledge are promptly factored into the technical considerations which form a part of the foundation for staff consideration of a reduced emergency planning zone radius.

51. Reference 1, page 3-7, paragraph 5) references both high and low level sump alarms. What is a sump low level alarm?

52. Page 3-7 contains a discussion of vault behavior in response to RHR system breaks. The emphasis is upon loss of equipment due to flooding. What consideration has been given to breaks which are small enough that the vault is not flooded, but there is a significant thermal energy release that may impact equipment operation? Please include consideration that enough energy may be released to activate the fusible links in the ventilation system, thereby terminating ventilation and indirectly causing failure to pumps due to overheating of pump motors, and that this could occur at a time earlier than might occur due to flooding.

53. Reference is made on page 3-7 to the RHR system crosstie line and RHR system response due to flow in this line as well as in the miniflow bypass lines. The conclusion is drawn that the RHR system pressure will tend to be uniform as a result. Are flow conditions such that this is realistic?

What is the impact of this assumption on conclusions pertinent to the discussion?

54. The authors conclude on page 3-9 that presence of water in the reactor cavity will decrease (significantly?) the revaporization of fission products from RCS and perhaps RHR surfaces. We anticipate that a significant quantity of heat producing radioisotopes will remain in the wreckage of the reactor vessel, and this may be effective in heating whatever gases or vapor are flowing toward the break. Has this been investigated?

55. What is the justification for the statement on page 3-10 that the first sign of trouble will be pressurizer low level or low pressure alarms? We suspect a number of other indicators may be first, such as abnormal indications from the PRT or even a smoke alarm.

56. There have been a number of indications (prior to and including page 3-11) that containment spray may be actuated due to RHR relief valve release into containment. What is the justification for this conclusion? Include the effect of containment heat sinks and containment cooler operation in the response.

57. The statement on page 3-11 that "As soon as the pumps begin to produce flow to the RCS, valves in the miniflow lines close and all RHR pump flow is injected into the reactor vessel via the RHR cold leg injection lines" is not correct. The sensors are not located at the RCS to detect flow at that location. Further, one is postulating a break in the RHR system, and a significant portion of the pump flow may never reach the RCS (as it stated in a later paragraph).

58. The last paragraph on page 3-11 contains a number of timing of event statements. Please provide justification of each. Plots of plant behavior showing suitable parameters and indicating the event points are sufficient for most. Operator response information, in addition to RCS parameter information, is necessary to substantiate the statement that RCPs will be tripped within about 21 seconds of break initiation.

59. An item under consideration for advanced nuclear power plants is the ability to monitor pressure on the low pressure side of check valves. This could provide early warning of check valve leaks and would provide monitoring capability to help assure check valves were operating properly. The same monitoring capability with respect to RHR suction line valves could identify if individual valves were mispositioned or malfunctioning. Would such a system for Seabrook be of significant benefit in reducing risk in a reduced size emergency planning zone?

60. Please elaborate on the page 3-23 list of actions an operator can take to mitigate the accident. This list appears to be short. Include identification of what has been incorporated into operator training and procedures at Seabrook.

61. What is the frequency of failures in the pipe tunnel that is mentioned on page 3-23, and which led the authors to conclude they are very low?

62. Page 3-27 references situations where the combined sump pump capacity is sufficient to remove leaks and keep the vaults from flooding. In these cases, the RHR, SI, and CS pumps are assumed not to be impacted by flooding. What consideration was given to failure of one (or both) sump pumps?

63. What is the maximum flow rate that can be injected into the RCP pump seals? (Of potential interest since it may be an alternate path for injection into the RCS.)

64. Shutting an RHR system crosstie valve is identified on page 3-35 as an action to help isolate a LOCA outside containment involving the RHR/SI systems. Has a careful evaluation of these systems been performed to assess isolation strategy? If so, are procedures in place at Seabrook Station which reflect the work?

65. Relative water levels in the RHR vaults and the RCS are mentioned on pages 3-35 and 3-36. What are the water volumes in these regions as a function of elevation? (Of particular interest is the level at the top of the core and at the elevation of the hot leg connections to the RHR.)

66. What is the justification for the statement on page 3-36 that the water level in the vaults will be approximately the same as that in the RCS? (We do not agree because of the potential that pressure in the vaults and containment are not the same, and water temperature in the two locations may differ.)

67. Page 3-37 contains the wording "End state DLOC contains sequences in which the interfacing LOCA has been terminated, and the ECCS has been degraded (D) (RHR or SI pumps have failed)....The point estimate frequency of DLCO is 4.0×10^{-7} per year. The additional failures required to achieve core melt would lower this frequency by a least one order of magnitude." What is the justification for this conclusion? (We have already lost a portion or all of the ability to inject water into the RCS via the usual paths.)

68. The bottom of page 3-37 contains a statement to the effect that failure of one charging pump will lead to core melt. Why is this the case? Our perception is that sufficient flow might be provided by alternate means to keep the core covered, such as use of the remaining two charging pumps, and perhaps the reactor makeup water pumps).

69. What is to be the status of the "temporary" 34.5 kV power lines which are identified on page 3-45?

70. What is to be the status of the mobile power supplies which are identified on page 3-46?

71. What capability has been provided to connect external pumps as identified in the second and third paragraphs of page 3-46? (This was briefly mentioned on page 3-48.) Use of a pump to simply inject water into containment via the sprays on a short term basis (no recirculation) does not appear to be identified. Has this been considered?

72. Page 3-46 identifies a number of possibilities for recovery of various safety functions. Are there specific plans? If so, please provide them.

73. There have been several references to purchase of a mobile electric generator by pooled resources on the pages prior to page 3-49. What is the likelihood that such a generator would be needed by several plants at the same time, and hence might not be available to Seabrook Station when needed? Similarly, where is the generator to be stored, and how is it to be transported to Seabrook? Include consideration of post seismic and post severe storm conditions in the response.

74. A tacit assumption appears to be incorporated into References 1 and 2 that check valves are always closed. In reality, many check valves require a (substantial) reverse flow to force them to close, and they additionally often require a significant reverse pressure to keep them closed. It this the case for any of the valves of interest here? If so, please discuss the implications. If not, what is the justification for the conclusion?

75. In the description of RHR pressure boundary failure modes it is stated that the maximum value of stresses due to pressurization to 2250 psia in the limiting RHR piping are approaching the yield stress and the stresses in other metallic components are at a small fraction of their respective yield stresses. Describe the analyses conducted to support this conclusion and provide a summary of the pertinent results. In addition, clarify whether the pressure loading has been applied as a dynamic pulse coupled with corrosion degradation effects (such as heat exchanger tube embrittlement). If these effects have been considered, describe the analyses and the dynamic loads. If not, provide the bases for not considering these effects.