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

October 5, 1992

NOTE TO: Jack Duncan, GE

FROM: Glenn Kelly, SPSB, DSSA, NRR

SUBJECT: QUESTIONS RELATED TO GE ABWR PRA SUBMITTALS (JUNE AND JULY 1992)

I have enclosed a list of questions (Enclosure 1) dealing with the reliability assurance program, the ABWR seismic margins analysis, the RWCU, and the requantification of the ABWR PRA. Please contact me if they require any clarification.

At this time we are working on a better description of the staff's expectations regarding a seismic margins analysis for the ABWR PRA. We expect to complete this guidance shortly.

I have also enclosed a draft set of questions (Enclosure 2) that we are still considering regarding the DHR reliability study. It has been provided for your information only, and you need not take any steps to respond to it until the questions are final.

Enclosures: as stated

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ENCLOSURE 1

SPSB QUESTIONS ON THE ABWR PRA
October 5, 1992

* Reliability Assurance Program - (1) GE has indicated an intent not to duplicate requests for inclusion of SSCs into RAP if they are already there from deterministic insights. The staff considers it important to identify to the COL applicant the insights from the PRA regarding SSCs to be included in RAP. If the COL applicant is not provided with PRA-based SSC insights, the applicant may design and perform periodic tests that do not fully challenge the SSC in areas necessary to assure it will perform during the events postulated in the PRA (and for which it is credited). The staff believes it is necessary for GE to make a complete listing of SSCs to be included in RAP based on the insights from the ABWR PRA. (2) The staff continues to await a listing of reliability targets in table form for systems (train or system-wide level) and components identified for inclusion in the RAP. (3) At a function level, there are several important systems that act as redundant components that, in the aggregate, provide an important safety function, but do not individually have a high importance (as determined by importance measures). Expand your SSC list to include a single train of the following systems: HPCF, RWCU, RHR, LPFL, RSW, and electrical ac. A walkdown inspection of these systems should be performed periodically (perhaps on a yearly basis) to verify that no unnecessary common cause failures exist between the independent divisions. A checklist could be provided to facilitate and guide the inspections. (4) Add a COL action item to have the COL applicant provide a listing of individual barriers that should be included in the RAP for fire. These barriers should include doors, separation walls/floors between safety divisions, and penetrations between divisions. (5) Expand RAP to include a requirement that the COL applicant review and exercise annually the emergency operation procedure for operating the Remote Shutdown Panels and manually operating RCIC from outside of the control room in emergency situations. (6) Expand RAP to include fire dampers in the HVAC system.

* Seismic Margins - (1) Explain why for sequences 15, 16, 17, and 18 the event trees do not consider the possibility of failure of RHR heat exchanger integrity. (2) Provide the fragility data for the condensate injection (V2). (3) Explain why no seismic fault trees or fragility values were provided for depressurization, level and pressure control, and inhibit ADS. (4) Discuss the seismic capability of the isolation valves and their controllers in the OPS. What prevents the OPS line from becoming kinked or obstructed during a seismic event? (5) Your seismic analysis assumes that an earthquake would not prevent the rupture disk(s) from opening. Certainly, failure closed of one of the isolation valves would fail the COPS. So would pressurization of the volume between the rupture discs. Provide a more complete discussion of why Class II sequences do not need to be considered for seismic events. (6) Discuss whether LPCF, RCIC, and HPCF are capable of pumping saturated fluids at 365°F. If not, what is the basis of taking credit for them in Class II sequences, if the rupture disc has not opened (due for example to a crimp in the line or failure closed of an isolation valve) and the pressure in containment has increased to containment

ultimate. (7) Discuss the implications of seismic failure of the SRV discharge line, with or without sprays available. Would the ADS be compromised? Is containment challenged? Are the SRV discharge lines modeled as part of the depressurization system? If not, why not? (8) Are "p" sequences to be considered as large releases, since they drain the suppression pool? If not, how should they be considered and why? (9) On page 8 of your June 26, 1992 fax on Seismic Margins Analysis, you should include SRV discharge line failure under number 9 (Depressurization). GE's ex-containment LOCA submittal indicated that these lines had a HCLPF of 0.72g.

* RWCU - (1) The SSAR needs to be modified to include a requirement that the COL applicant demonstrate that the proposed changes in the RWCU operating temperatures during severe accidents is acceptable during emergency use. (2) In the event of a high pressure LOCA or transient where the RWCU is needed to supply decay heat removal (i.e., RHR has failed), discuss when and if containment isolation would automatically occur and how, if at all, it would affect the use of the RWCU.

* PRA Requantification - Based on the LOOP and SBO event tree (Figure 19D.4-4), the sequence involving LOOP, followed by successful scram and recovery of offsite power within 30 minutes, has a frequency of 5.79E-2/yr. In GE's original PRA submittal, this sequence was transferred to the Reactor Shutdown event tree (Figure 19D.4-1). The staff has recommended that this sequence be transferred to the Isolation/Loss of FW event tree instead. In the revised submittal (May 21, 1992), this sequence was neither transferred to the Reactor Shutdown tree nor the Isolation/Loss of FW tree. Please explain this discrepancy.

ENCLOSURE 2

ABWR DECAY HEAT REMOVAL RELIABILITY

(1) The ABWR Shutdown Risk Evaluation (SRE) stresses the importance of having at least one offsite power, one diesel generator, and the gas turbine available at all times. This string of systems has a 3E-6 per week failure probability. (a) What is the basis for assuming that the gas turbine will be available, given it is not covered by Technical Specifications? (b) What does GE recommend a utility applicant do if the gas turbine or one of the other systems in the "string" becomes unavailable? (c) During a shutdown of longer than one week, what assures that this "string" of equipment is actually available?

(2) It appears that maintenance of the suppression pool was not modeled in the SRE. (a) Explain how the need for a utility to drain down the suppression pool when in modes other than full power is modeled in the SRE. (b) Provide the COL applicant with guidance on when it is appropriate from a risk perspective to drain down the suppression pool. (c) Indicate specifically what GE is recommending, if anything, to limit the potential risk during the period the suppression pool is lowered.

Although the SRE did not model suppression pool maintenance/failure, the draft NSAC study on shutdown risk at Grand Gulf reported that medium and large LOCAs overwhelmingly dominated the core damage risk during the RF04 outage. Twice during this outage, Grand Gulf entered a plant configuration in which the upper containment and suppression pools were drained. LOCAs during periods with the reactor cavity and suppression pool drained could cause failure of all ECCS. (d) Address the extent to which loss of the suppression pool influences the conditional core damage frequency estimates during modes other than full power and discuss potential operator errors that could exacerbate or mitigate this event.

(3) The staff finds the use of a 1E-4 screening value too coarse to preclude large releases occurring with a frequency higher than 1E-6 per year. (Note that all core melts occurring with primary containment open (e.g., when in mode 6) are considered to be large releases as fission products will not be scrubbed.) Revise the SRE using a screening value of 1E-5 or lower. The staff considers 1E-5 to be an acceptable screening value since when it is combined with an assumed loss of decay heat removal train frequency of 0.1 per year, it leads to a large release frequency of no higher than 1E-6 per year.

(4) GE should make the assumed failure frequency of a single decay heat removal train (i.e., 0.1 per year) a RAP item and reliability target.

(5) Provide cutsets for the dominant sequences.

(6) Justify the assumed maintenance intervals in the SRE during shutdown. Operating experience indicates that operating plants have longer equipment outages during modes other than full power than assumed in the SRE. For example, GE assumed a 10 percent likelihood that DHR train B would be

unavailable due to maintenance for shutdown cooling. However, the draft NSAC shutdown risk report concluded that, during Grand Gulf's RF04 outage, train A of DHR was unavailable for nearly 50 percent of the time due to train A divisional maintenance.

(7) GE assumes that protection against floods is afforded during modes other than full power by keeping the water-tight doors to one safety division closed. As an example, Divisions B and C could have their doors at the -8200 mm level open, but Division A doors would have to be shut. Then a flood occurring in B or C should only affect these divisions, and Division A should only be subject to random failures. (a) What are the consequences of a flood beginning in Division A instead of B or C? It is the staff's understanding that the water-tight doors are designed to open out under the head of water, allowing water to spill into the hallway common to the safety divisions. Would the water-tight doors of Division A open as they are designed and flood Divisions B and C too? (b) If the flood were to occur in Division B or C, what assures that the equipment in Division A is available, particularly since GE's analysis assumes one train of ac power is available? (c) How are the insights from the internal flood analysis tied into the ABWR shutdown T.S.? (d) It is typical during shutdown operations for there to be increased wood, plastic, paper, styrofoam, and other floatable objects in the reactor, turbine, and control buildings. Describe how your internal flooding analysis has considered the increased possibility of drains becoming plugged or sump pumps becoming disabled from floating debris. (e) GE states on page 38 of the SRE that "an analysis has been completed ... no more than one safety division would be affected by water damage from any postulated flood." This statement contradicts other information GE has supplied the staff on internal floods. Please explain.

Joint w/ Plant Syst Brnch

(8) A probabilistic analysis of the risk of fire during modes other than full power was not included in the SRE. Provide a systematic, quantitative analysis of fires during shutdown.

(9) In the SRE GE discounts the differences between the three divisions of the RHR system as being minor. However, division A of the RHR system cannot provide cooling to the spent fuel pool. Therefore, under conditions in which the RHR system is cooling one-third of the core (or more) in the spent fuel pool, only two divisions of RHR are available for fuel pool cooling. This condition occurs every refueling outage. Because of the unique dependence of the ABWR fuel pool cooling system on the RHR system during normal refueling operations, GE should provide a more thorough evaluation of the impact of loss of cooling to the fuel, whether it be in the reactor vessel or in the spent fuel pool.

(10) Because of the potential severity of a failure of the seals used during RIP pump and impeller replacement, the unsubstantiated statement that "the probability of a large leak through this path is small" is inadequate. Provide an analysis that addresses the probability of a LOCA through the impeller shaft nozzle. This analysis should address both mechanical failures and the potential for procedural errors in performing RIP maintenance (e.g., incorrect sequence of, or missing, procedural steps).

Joint w/ ISB

(11) In operating BWRs, events have occurred during shutdown in which RHR system misalignment has resulted in partial draining of the reactor vessel. The ABWR design has incorporated features to insure that the core cannot become uncovered directly as a result of RHR misalignment (e.g., the suction nozzles are above the core). However, such a misalignment could result in a common mode failure of the available RHR divisions (including shutdown cooling and core flooding) if pump damage occurs before the situation is corrected. It is not clear that the SRE has appropriately considered this. While an event does appear in the RHR (shutdown cooling) fault trees for low RPV level (P=1E-6, it is not clear if this event is meant to include low level due to RHR system misalignment), clearly pump failure as a result of low level is not considered in RHR core flooding mode. GE should defend not analyzing this common mode failure or should modify the SRE to include its analysis.

(12) AC power requirements described on pages 69 and 73 of the SRE seem to imply that minimal sets of equipment that include two divisions of RHR in the shutdown cooling mode may require that all three diesel generators be available, since the redundant division will require two ac power divisions and the operating division will require one. The first five minimal sets listed on Table 19Q7.2 all require three ac power divisions. This requirement does not appear correct. GE should verify these minimum sets to verify that the support system requirements associated with the identified configurations are realistic.

(13) In the shutdown safety issues section of the SRE (Section 19Q.10), GE discusses delaying DHR maintenance until heat loads are reduced or the core has been offloaded into the spent fuel pool. Since DHR system is required to provide spent fuel pool cooling for at least the first 21 days of shutdown, the overriding reason to delay maintenance is the reduced heat loads. However, only two DHR divisions are capable of cooling the fuel pool. The staff recommends modifying the discussion to more accurately reflect the reasons for delaying maintenance.

Verbal

(14) In the event trees in the SRE, GE apparently defined success as any sequence where systems are available to perform either the injection or heat removal functions. In part this appears to be the result of the assumption that a 24 hour mission time for all events during shutdown is sufficient for the analysis and boiling to the containment is an acceptable heat removal mechanism even if no containment heat removal mechanisms are available. Provide justification for the decision that only either injection or heat removal is needed.

(15) The success criteria in Section 19Q7 and the corresponding event trees include several paths. One path is direct decay heat removal from the reactor vessel using closed loop cooling like DHR or reactor water cleanup (CUW). This path requires that level, as measured in the downcomer, is sufficiently high to allow adequate recirculation between the downcomer region and the core. (a) Either justify why level control was not addressed in the event trees or modify the event trees.

*Only in mode 4 & flooded. KI's assuming mode 5 flooding + reason.

A second path is via coolant makeup using either HPCF, LPCF, condensate pumps, or CRD. For this path, level control was modeled, but the source of energy

removal from containment was not addressed. It appears that the analysis assumes that injection systems would be used to pass two phase or steam flow through the SRVs back to the suppression pool. This method of core cooling assumes that the SRVs are available, which is not true for the entire outage (To flood the refueling cavity, the vessel head is removed and the steam lines are plugged, resulting in the SRVs being unavailable). This mode of energy removal using the SRVs would seem to require some type of containment cooling to prevent containment overpressurization. (b) GE does not appear to have considered SRV availability in its event trees. Justify this omission or modify your trees. (c) ~~Does GE take credit for use of the OPS in these circumstances? (d) Can the OPS be out for maintenance? If so, how is this~~ modeled in the SRE? (e) During refueling when the head is off the vessel and the steam lines are plugged, core cooling involves having steam or two-phase flow exit the vessel directly. Using HPCF, LPCF, etc. for core cooling would seem to require suppression pool makeup to maintain suppression pool inventory. It does not appear that the event trees consider the need for inventory makeup in this situation. Please explain or modify the event trees.

(16) The staff has identified human errors (HEs) as perhaps the single most significant contributor to shutdown risk. (a) The staff requests GE to provide an estimate of the contribution of HEs to core damage frequency in modes other than full power. This can be done by extending the sensitivity analysis to identify and characterize the leading HEs. (b) GE should discuss the tolerance of the ABWR to HEs during modes other than full power. (c) GE should provide COL applicants with guidance for modes other than full power regarding administrative controls, procedures, and Technical Specifications based on assumptions and insights from the ABWR SRE.

(17) GE's analysis assumes that steaming is an acceptable method of removing decay heat if there is makeup available. (a) Where would the steam go in modes 4 and 5? (b) What equipment would be subject to this steam environment? (c) Would the steam fail any equipment needed to keep the core cool? (4) What areas, besides secondary containment, would the steam prevent operators from entering?

Suppression pool thru SRVs

secondary cont to refueling floor.

DRAFT

October 29, 1992

Sent by fax to J Duncan 11/2/92

Docket No. 52-001

Mr. Patrick W. Marriott, Manager
Licensing & Consulting Services
GE Nuclear Energy
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Marriott:

SUBJECT: DECAY HEAT REMOVAL RELIABILITY STUDY FOR THE GE NUCLEAR ENERGY ADVANCED BOILING WATER REACTOR (ABWR)

Enclosed is a list of questions concerning the U.S. Nuclear Regulatory Commission staff's reliability study and evaluation of internal floods and fires during modes other than full power. The questions should not be regarded as a request for new information, but, rather, to provide clarification and enhance the staff's understanding of the ABWR design in these areas.

Sincerely,

(Original signed by)
Robert C. Pierson, Director
Standardization Project Directorate
Associate Directorate for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

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Enclosure

ABWR DECAY HEAT REMOVAL RELIABILITY

(1) The ABWR Shutdown Risk Evaluation (SRE) stresses the importance of having at least one offsite power, one diesel generator, and the gas turbine available at all times. This string of systems has a 3E-6 per week failure probability. (a) What is the basis for assuming that the gas turbine will be available, given it is not covered by Technical Specifications? (b) What does GE recommend a utility applicant do if the gas turbine or one of the other systems in the "string" becomes unavailable? (c) During a shutdown of longer than one week, what assures that this "string" of equipment is actually available?

(2) It appears that maintenance of the suppression pool was not modeled in the SRE. (a) Explain how the need for a utility to drain down the suppression pool when in modes other than full power is modeled in the SRE. (b) Provide the COL applicant with guidance on when it is appropriate from a risk perspective to drain down the suppression pool. (c) Indicate specifically what GE is recommending, if anything, to limit the potential risk during the period the suppression pool is lowered.

Although the SRE did not model suppression pool maintenance/failure, the draft NSAC study on shutdown risk at Grand Gulf reported that medium and large LOCAs overwhelmingly dominated the core damage risk during the RF04 outage. Twice during this outage, Grand Gulf entered a plant configuration in which the upper containment and suppression pools were drained. LOCAs during periods with the reactor cavity and suppression pool drained could cause failure of all ECCS. (d) Address the extent to which loss of the suppression pool influences the conditional core damage frequency estimates during modes other than full power and discuss potential operator errors that could exacerbate or mitigate this event.

(3) The staff finds the use of a 1E-4 screening value too coarse to preclude large releases occurring with a frequency higher than 1E-6 per year. (Note that all core melts occurring with primary containment open (e.g., when in mode 6) are considered to be large releases as fission products will not be scrubbed.) Revise the SRE using a screening value of 1E-5 or lower. The staff considers 1E-5 to be an acceptable screening value since when it is combined with an assumed loss of decay heat removal train frequency of 0.1 per year, it leads to a large release frequency of no higher than 1E-6 per year. (4) GE should make the assumed failure frequency of a single decay heat removal train (i.e., 0.1 per year) a RAP item and reliability target.

(5) Provide cutsets for the dominant sequences.

(6) Justify the assumed maintenance intervals in the SRE during shutdown. Operating experience indicates that operating plants have longer equipment outages during modes other than full power than assumed in the SRE. For example, GE assumed a 10 percent likelihood that DHR train B would be unavailable due to maintenance for shutdown cooling. However, the draft NSAC shutdown risk report concluded that, during Grand Gulf's RF04 outage, train A of DHR was unavailable for nearly 50 percent of the time due to train A divisional maintenance.

(7) AC power requirements described on pages 69 and 73 of the SRE seem to imply that if two divisions of RHR in the shutdown cooling mode are required to be operable, it would require that all three diesel generators are available. This is because an isolation valve in the suction line in shutdown cooling train A is powered by a power source from another division. The same is true for trains B and C. So for trains A and B to be operable, power sources from divisions A, B, and C would need to be operable. Notice that the first five minimal sets listed on Table 19Q7.2 all require three ac power divisions. This design appears on the surface to potentially increase the estimated core damage frequency. Has GE determined that the threat from a potentially unisolable line break is more significant than having to have three divisions of electrical power operable during modes other than full power? GE should verify these minimum sets to verify that the support system requirements associated with the identified configurations are correct and should clarify the need to have SDC suction line isolation valves powered as they appear in the current design.

(8) The success criteria in Section 19Q7 and the corresponding event trees include several paths. One path is direct decay heat removal from the reactor vessel using closed loop cooling like DHR or reactor water cleanup (CUW). This path requires that level, as measured in the downcomer, is sufficiently high to allow adequate recirculation between the downcomer region and the core. (a) Either justify why level control was not addressed in the event trees or modify the event trees.

A second path is via coolant makeup using either HPCF, LPCF, condensate pumps, or CRD. For this path, level control was modeled, but the source of energy removal from containment was not addressed. It appears that the analysis assumes that injection systems would be used to pass two phase or steam flow through the SRVs back to the suppression pool. This method of core cooling assumes that the SRVs are available, which is not true for the entire outage (To flood the refueling cavity, the vessel head is removed and the steam lines are plugged, resulting in the SRVs being unavailable). This mode of energy removal using the SRVs would seem to require some type of containment cooling to prevent containment overpressurization. (b) GE does not appear to have considered SRV unavailability in its event trees in mode 4. Justify this omission or modify your trees. (c) During refueling when the head is off the vessel and the steam lines are plugged, core cooling involves having steam or two-phase flow exit the vessel directly. Using HPCF, LPCF, etc. for core cooling would seem to require suppression pool makeup to maintain suppression pool inventory. It does not appear that the event trees consider the need for inventory makeup in this situation. Please explain or modify the event trees.

(9) The staff has identified human errors (HEs) as perhaps the single most significant contributor to shutdown risk. (a) The staff requests GE to provide an estimate of the contribution of HEs to core damage frequency in modes other than full power. This can be done by extending the sensitivity analysis to identify and characterize the leading HEs. (b) GE should discuss the tolerance of the ABWR to HEs during modes other than full power. (c) GE should provide COL applicants with guidance for modes other than full power regarding administrative controls, procedures, and Technical Specifications based on assumptions and insights from the ABWR SRE.

(10) This question is asked in parallel with the Reactor Systems Branch. GE's

analysis assumes that steaming is an acceptable method of removing decay heat if there is makeup available. (a) Where would the steam go in mode 5? (b) What equipment would be subject to this steam environment? (c) Would the steam fail any equipment needed to keep the core cool? (4) What areas, besides secondary containment, would the steam prevent operators from entering?

(11) This question is asked in parallel with the Plant Systems Branch. GE assumes that protection against floods is afforded during modes other than full power by keeping the water-tight doors to one safety division closed. As an example, Divisions B and C could have their doors at the -8200 mm level open, but Division A doors would have to be shut. Then a flood occurring in B or C should only affect these divisions, and Division A should only be subject to random failures. (a) What are the consequences of a flood beginning in Division A instead of B or C? It is the staff's understanding that the water-tight doors are designed to open out under the head of water, allowing water to spill into the hallway common to the safety divisions. Would the water-tight doors of Division A open as they are designed and flood Divisions B and C too? (b) If the flood were to occur in Division B or C, what assures that the equipment in Division A is available, particularly since GE's analysis assumes one train of ac power is available? (c) How are the insights from the internal flood analysis tied into the ABWR shutdown T.S.? (d) It is typical during shutdown operations for there to be increased wood, plastic, paper, styrofoam, and other floatable objects in the reactor, turbine, and control buildings. Describe how your internal flooding analysis has considered the increased possibility of drains becoming plugged or sump pumps becoming disabled from floating debris. (e) GE states on page 38 of the SRE that "an analysis has been completed ... no more than one safety division would be affected by water damage from any postulated flood." This statement contradicts other information GE has supplied the staff on internal floods. Please explain.

(12) This question is asked in parallel with Plant Systems Branch. A systematic analysis of the risk of fire during modes other than full power was not included in the SRE. Provide a systematic, quantitative analysis of fires during shutdown similar to that requested for internal floods in question 11 above.

(13) This question is asked in parallel with the Reactor Systems Branch. Because of the potential severity of a failure of the seals used during RIP pump and impeller replacement, the unsubstantiated statement that "the probability of a large leak through this path is small" is inadequate. Provide an analysis that addresses the probability of a LOCA through the impeller shaft nozzle. This analysis should address both mechanical failures and the potential for procedural errors in performing RIP maintenance (e.g., incorrect sequence of, or missing, procedural steps).