GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

March 4, 1993

Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: **Submittal Supporting Accelerated ABWR Review Schedule - ABWR Probabilistic Flooding Analysis**

Dear Chet:

Enclosed are responses to questions on the ABWR probabilistic flooding analysis discussed with Glenn Kelly on February 22, 1993 and documented in an NRC letter dated February 25, 1993.

Please provide a copy of this transmittal to Glenn Kelly.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Jack Duncan (GE)
Norman Fletcher (DOE)
Art McSherry (GE)

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InterOffice Memo

To: J.N. Fox

From: Art McSherry *Art McSherry*

Date: March 3, 1993

Subject: Responses to NRC Questions on ABWR Flooding

CC: J.D. Duncan, S. Visweswaran

The following questions on the ABWR Probabilistic Flooding analysis were discussed with the NRC (Mr. Glenn Kelly) on February 22, 1993. The requested information was documented by an NRC letter dated February 25, 1993. The following is a discussion of the questions from the February 25, 1993 letter.

Question 1 - Explain the difference in reality and in the assumptions in the ABWR PRA between a fire door and a watertight door.

Response 1 - From a flooding perspective, the main difference between fire doors and watertight doors is that fire doors may inhibit or stop flood progression but watertight doors always do (if closed). Therefore, when analyzing fire door impacts on flooding, the worst case is assumed (i.e., if leaking results in water being diverted from safe shutdown equipment, the door is assumed to not leak; if no leaking results in protecting safe shutdown equipment from flooding, it is assumed to leak). A good example of this is flooding in the reactor building corridor floor B3F. If the fire doors in the corridor were to leak, the entire corridor volume would be available to contain potential flood waters. In this case, the doors were assumed to not leak which minimized the available room volume and increased the flood height in the room.

Question 2 - When a watertight door is closed but not dogged, will it alarm in the control room?

Response 2 - The current level of detail in the ABWR reactor building design does not address the specifics of door alarms. Operating plant designs do not indicate if watertight doors are dogged or not, only that the door is closed. Administrative procedures require that a once per shift walkdown of the plant be completed and any undogged watertight door will be considered a reportable event. Experience at operating plants has found this to be adequate to ensure that watertight doors remain dogged when required.

The SSAR will be changed to reflect this requirement.

Question 3 - Explain how CCF of operators leaving all three ECCS watertight doors open was considered. Explain how operator error is considered in flooding analysis.

Response 3 - The study assumed that a watertight door to an ECCS room could fail (leak or be open) but the probability of all three watertight doors being open at the same time was negligible. Procedures will exist to instruct operators to ensure that at least one watertight door is closed at all times and if flooding were to occur in a room with a closed watertight door that the other two watertight doors would be checked closed before opening the door to the flooding room.

This is discussed in Section 19R.4.2.4 of the ABWR SSAR.

Question 4 - Provide a better description of the worst flood inside of containment as estimated by the flooding PRA.

Response 4 - Section 19R.5.5 describes the reactor building flooding scenario. Figure 19R.5-6 shows the reactor building flooding in secondary containment event tree. The worst case scenario is sequence number 7 which has a CDF of 2.0E-10 per reactor year. For this sequence, a break in a fire water standpipe or line from the CST results in the sump level switches detecting the flood (DET) but the operator does not respond and manually isolate the flood (OPACT). The sump pump capacity is exceeded (SUMPP) and the overfill lines to the corridor are clogged (OFLC). This results in flooding of all three electrical rooms on floor B1F and loss of all AC powered makeup systems. Core damage results when the operator fails to depressurize and use the AC independent water addition system to inject into the RPV. The operator failure rate for this sequence is .1 due to the short time available (<30 minutes) to isolate the flood before damage to the electrical equipment.

The SSAR will be changed to include the above scenario description.

Question 5 - Provide explanation of flooding ET top events.

Response 5 - The ET headings are explained below.

ET	HEADING	EXPLANATION
Turbine Building Flooding (Low UHS)	CWS Pipe Break in Turbine Building	Break in CWS causes flooding in the turbine building
	Level Switches Detect Flood	Level switches in the condenser bay detect water from the flood.
	All three pumps trip (pump breakers open)	CWS pump breakers open terminating the flood since the UHS is low.
	All three MO Valves close	All three valves in the CWS must close because the break could be in the header that all three pumps feed.
	Truck entrance door opens.	The truck entrance door at grade level is not watertight and it is expected that it would leak due to the presence of the water.
	NC Watertight CB Excess Dr Closed.	The door from the turbine building to the service building access tunnel (entrance to the control building) is closed to prevent control building flooding.

	Reactor brought to a safe shutdown.	The reactor is safely shutdown using equipment not damaged by the flood. This heading is essentially the same as a turbine trip without bypass event with the additional loss of condensate. See the answer to question 6 for additional information.
Turbine Building Flooding (High UHS)	Headings are the same as the previous ET except that no credit is taken for pump trip due to the high UHS	
Control Building Flooding	RSW Pipe Break in Control Building	A break in the RSW system piping inside the RSW/RCW room of the control building upstream of the isolation valve.
	Lower level sensors at .15m detect RSW room Flooding	The 2/4 water level sensors in the room detect the flood water and send a signal to the control room to alert the operator.
	Operator acts to isolate flooding.	Operator receives the alarm and either trips the affected RSW pump or closes appropriate valves in the loop.

	Upper level sensors at .8m detect RSW/RCW room flooding	The next higher level sensors detect the flood and send a signal to the control room and automatically trip the RSW pump and close isolation valves in the affected loop.
	Level sensors in the next two rooms detect RSW/RCW flooding	If the sensors in the first room fail or the operator does not respond, the first RSW room will overflow into the other two RSW rooms and the sensors in those rooms will detect the flood.
	Operator acts to isolate flooding.	Based on alarms from the sensors in the other rooms, the operator can determine that flooding exists and isolate the flood.
	Automatic flooding isolation.	RSW pump trip or MOV closure will terminate the flood. See response to question 8 below for additional information.
	Reactor brought to a safe shutdown condition.	Using features not damaged by the flood, the plant is successfully shutdown. See response to question 6 for additional details.
Reactor building flooding in ECCS room.	Pipe break in ECCS room.	Double ended shear of line from CST or suppression pool upstream from the isolation valve.

	Sump level switches detect flood.	Sump water level alarms in the control room.
	Water in the corridor.	Watertight doors fail to contain flood water in the ECCS room and water flows into the corridor.
	Water in next division ECCS room.	Watertight door in other division fails enabling water to enter from the corridor into the ECCS room and results in loss of two ECCS divisions.
	Reactor brought to a safe shutdown.	Using equipment not damaged by the flood, the reactor is manually shutdown. The conditional probability of core damage is1.27E-6 for loss of one division and 2.68E-5 for loss of two divisions. See response to question 6 for additional information.
Reactor building flooding in corridor.	Pipe break in the corridor.	Break in fire water standpipe or line from suppression pool or CST causes flooding in the corridor.
	Level switches detect flood.	Sump level switches alarm in the control room.

	Operator acts to stop flooding.	Operator closes CST or suppression pool isolation valves from the control room.
	Water enters one ECCS room.	Watertight door to ECCS room fails resulting in loss of one ECCS division. CCF of all three doors is considered negligible. See response to question 3.
	Reactor brought to a safe condition.	Reactor shutdown using equipment not damaged by the flood. The conditional CDF is 1.0E-8 for all divisions available and 1.27E-6 for loss of one division. See response to question 6 for additional details.
Reactor building flooding in secondary containment.	Pipe break outside secondary containment.	Break in fire water standpipe or line from CST results in flooding on B1F.
	Flow switches detect flooding.	Sump level switches alarm in the control room.
	Operator isolates flooding.	Operator closes appropriate valves from the control room isolating the flood.
	Sump pumps capacity exceeded.	Sump pumps on B1F cannot keep up with the flood.

Overfill lines clogged.	Lines to B3F from sumps in B1F are clogged resulting in flooding of all three electrical rooms.
Reactor brought to a safe condition.	All makeup assumed lost except AC independent fire water. Operator action outside control room required to implement fire water addition.

Question 6 - Provide FT for final node that considers all other ways to keep core cooled. Explain which systems GE takes credit for on a sequence by sequence basis.

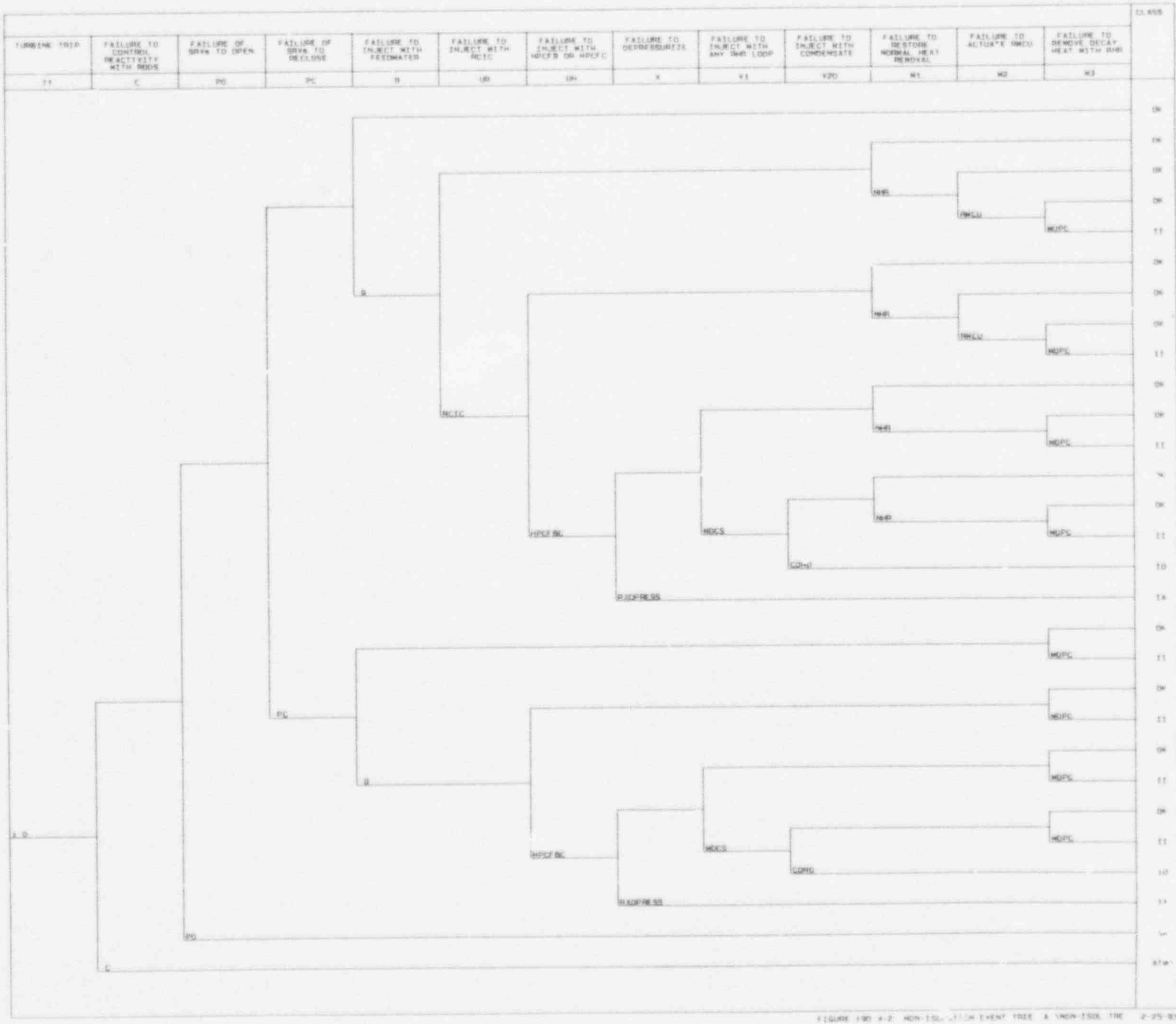
Response 6 - Attached as Figures 1 and 2 are the event trees that were used to determine the final nodes. Table 19Q.5-3 lists the final node probabilities. Conditional events 1,2, and 4 from Table 19Q.5-3 are all from the turbine trip without bypass event tree which is shown as Figure 1. The fault trees for RSW and power were modified to model loss of one and two divisions. 1.1E-8 assumes all divisions are available following a turbine trip without bypass. 1.27E-6 is for two divisions available and 2.68E-5 is for one division available. Figure 2 shows the event tree for loss of all RSW. The conditional probability is 6.38E-4. The fifth conditional probability in Table 19Q.5-3 is the CDF from the ABWR full power PRA turbine trip with bypass event tree assuming the initiating event frequency is 1. The last conditional probability is for loss of all three divisions of RSW and loss of the PCS. The .1 conditional probability is for the operator failure to implement the AC independent water addition system which would be the only makeup source available for this scenario.

Question 7 - Discuss credit taken for turbine building flood being mitigated by truck entrance door. Justify assuming 95% of time it will mitigate flood when curb at watertight door to support (service) building is only about four inches high.

Response 7 - The 0.05 unavailability is based on engineering judgment and the fact that the truck door is of a standard design for truck access (i.e., not a fire or watertight door) and will not present significant resistance to flood waters.

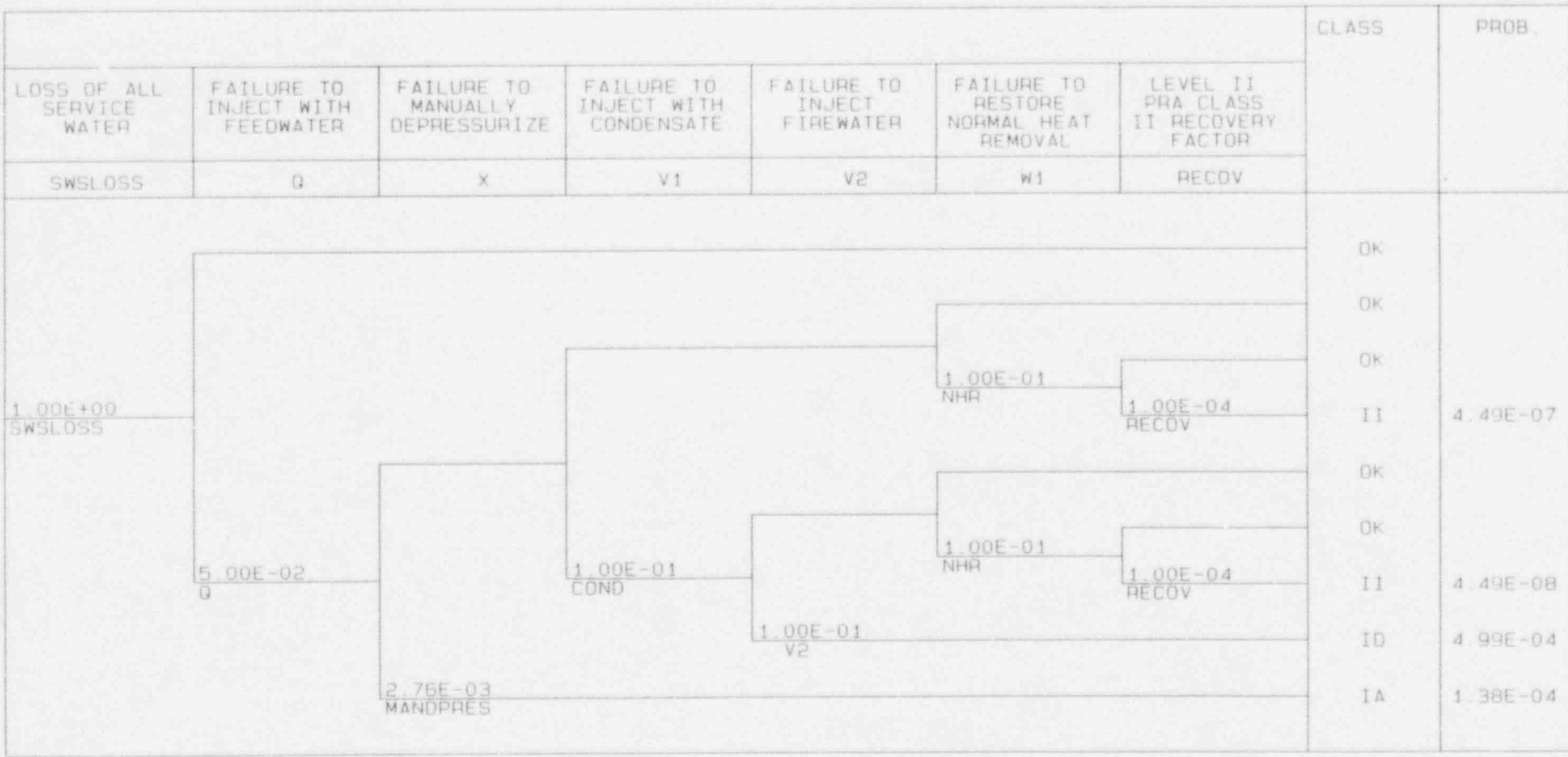
Question 8 - In Figure 19R.5-3 what is the basis for the (8)E-7 unavailability of flooding isolation.

Response 8 - The conditional probability of flooding isolation (8.0E-7) is the probability that the RSW isolation valves and pump trip logic will not function given that the water level sensors do actuate. The component unavailabilities are listed in Table 19R.5-2. Valve isolation is 4.0E-3. Pump trip is 1.0E-3. The combined failure probability of pump trip and valve closure takes into account the fact that only one of the two valves needs to close (see Figure 19R.4-2) to terminate the flood. Common cause failure of .1 for the valves to close are also considered. RSW pump trip or antisiphon valve failure is 1.0E-3 + 1.0E-3 = 2.0E-3. MOVs failure to close is 4.0E-3 per MOV and using a CCF of .1 results in failure of two MOVs being 4.0E-4. Therefore, the total failure probability is 2.0E-3 * 4.0E-4 = 8.0E-7.



TURBINE TRIP
TT
FAILURE TO CONTROL REACTIVITY WITH RODS
C
FAILURE OF SRVs TO OPEN
PO
FAILURE OF SRVs TO RECLOSE
PC
FAILURE TO INJECT WITH FEEDWATER
Q
FAILURE TO INJECT WITH RCIC
UR
FAILURE TO INJECT WITH HPCFB OR HPCFC
UH
FAILURE TO DEPRESSURIZE
X
FAILURE TO INJECT WITH ANY RHR LOOP
V1
FAILURE TO INJECT WITH CONDENSATE
V2D
FAILURE TO RESTORE NORMAL HEAT REMOVAL
W1
FAILURE TO ACTUATE RWCU
W2
FAILURE TO REMOVE DECAY HEAT WITH RHR
W3
CLASS
RCIC
HPCFBC
RXDEPRESS
NHR
RWCU
MDPC
MDCS
CDMD

FIGURE 1



LOSS OF ALL SERVICE WATER INITIATOR SWSLOSS2.TRE 5-30-92

FIGURE 2