ENERGY NORTHWEST

P.O. Box 968 ■ Richland, Washington 99352-0968

November 8, 2001
GO2-01-153

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **COLUMBIA GENERATING STATION, OPERATING LICENSE NPF-21 REQUEST FOR ADDITIONAL INFORMATION REGARDING THE UNISOLABLE DRAIN LINE BETWEEN THE CONTROL ROD DRIVE/CONDENSATE PUMP AND REACTOR CORE ISOLATION COOLING PUMP ROOMS**

References:
1) Letter, dated April 16, 2001, RL Webring (Energy Northwest), to NRC, "Request for Amendment, Unisolable Piping Run Between Control Rod Drive and Reactor Core Isolation Cooling Pump Rooms"
2) Letter, dated September 24, 2001, J Cushing (NRC) to JV Parrish (Energy Northwest), "Request for Additional Information (RAI) for the Columbia Generating Station (TAC NO. MB1777)"

Energy Northwest requested an amendment to Operating License NPF-21 regarding an unisolable piping run between the control rod drive and reactor core isolation cooling rooms (Reference 1). The NRC has requested additional information regarding the license amendment (Reference 2). The Energy Northwest response to the NRC request is attached.

Should you have any questions or desire additional information regarding this matter, please call me or Mr. RN Sherman at (509) 377-8616.

Respectfully,

D.W. Coleman

DW Coleman, Manager
Performance Assessment and Regulatory Programs
Mail Drop PE20

Attachment

cc: EW Merschoff - NRC-RIV
JS Cushing - NRC-NRR
NRC Sr. Resident Inspector - 988C
DL Williams – BPA-1399
TC Poindexter - Winston & Strawn

A075

NRC Question #1:

Please provide the risk analysis including assumptions used to determine that the increase in core damage frequency is less than 1E-10.

Energy Northwest Response:

Description

Energy Northwest estimated the impact on core damage frequency (CDF) from a potential flooding event combined with the unisolable Equipment Drain Radioactive (EDR) drain line between the Reactor Core Isolation Cooling (RCIC) pump room and EDR-SUMP-R5 in the Control Rod Drive/Condensate (CRD/COND) pump room. The RCIC pump room and the CRD/COND pump room are on the same elevation such that flooding in one area causes flooding in the other due to the unisolable EDR drain line. The condensate pumps in the CRD/COND pump room are part of the Condensate Storage and Transfer (CST) System, not the Main Condensate System.

Through wall pipe cracks, as defined in the required design basis flooding analysis, are the most probable pipe or equipment failure mechanism to initiate a flooding event. Through wall cracks that result in modest leakage flow rates allow relatively long times for operators to identify and isolate the source of the leak and to mitigate the consequences of the resultant flooding. Because of the relatively long times afforded for operator response, it is assumed that through wall cracks that result in modest leakage flow rates have a negligible impact on CDF and consequently, they are not included in the Columbia Generating Station (Columbia) Probabilistic Safety Assessment (PSA) flooding analysis.

The PSA flooding analysis estimates the impact of large pipe or equipment ruptures (as opposed to design basis pipe cracks). Large pipe or equipment ruptures have less probability of occurrence than through wall cracks, but are considered possible in any system. Large pipe or equipment ruptures result in relatively large flow rates that approach the maximum flow rate of the affected system pump. These ruptures allow relatively short times for operators to identify and isolate the source of the leak and to mitigate the consequences of the resultant flooding. It is therefore assumed that these ruptures could impact the CDF and thus, they are included in the Columbia PSA flooding analysis.

The Columbia PSA flooding analysis conservatively assumes RCIC system failure at approximately 6 inches water level on the RCIC pump room floor. Electrical conduit junctions at this elevation in the RCIC pump room carry electrical cables serving RCIC that are assumed to have cable splices. It is also assumed that submergence of these electrical conduit junctions containing spliced electrical cables will result in RCIC system failure. However, the PSA flooding analysis does not consider submergence of continuous insulated electrical cable a failure mechanism.

The Columbia PSA flooding analysis models manual plant shutdown and plant scram and thus, accounts for the potential impact of losing the CRD/COND pumps due to flooding.

Flooding initiated in the RCIC pump room partially flows to the CRC/COND pump room through the EDR drain line. The PSA flooding analysis does not take credit for water removal by the EDR sump pump. Partial flow of RCIC pump room floodwater to the CRD/COND pump room through the EDR drain line increases the total floor area subject to flooding and thereby slows the rate of rise in floodwater levels. This increases the time available to plant operators to identify and isolate the source of flooding and to mitigate potential damages from the flooding. Therefore, the unisolable EDR drain line slightly reduces the CDF for floods initiated in the RCIC pump room but this effect is small and not quantified.

Flooding initiated in the reactor building above the 422-foot elevation flows into the RCIC pump room by leaking through the marine style door R005, and by leaking past the floor plugs on the 471-foot elevation in the ceiling of the RCIC pump room. Most large pipe ruptures above the 422-foot elevation result in failure of the RCIC system. The EDR drain line delays the loss of RCIC by channeling a portion of the floodwater to the CRD/COND pump room. Therefore, the unisolable EDR drain line slightly reduces the CDF for floods initiated in the reactor building above the 422-foot elevation, but this effect is small and not quantified.

If the EDR drain line were isolated, then floods initiated in the CRD/COND pump room would result in a plant scram without impacting RCIC. However, with the unisolable EDR drain line, floods initiated in the CRD/COND pump room also result in concurrent flooding of the RCIC pump room. As a result, floods initiated in the CRD/COND pump room also result in loss of the RCIC system when the floodwater reaches approximately 6 inches on the RCIC pump room floor. Therefore, the unisolable EDR drain line increases the CDF due to floods initiated in the CRD/COND pump room. The impact of flooding initiated in the CRD/COND pump room is evaluated and described below.

Evaluation

Flood sources in the CRD/COND pump room that have the highest impact on CDF are the Division 1 Service Water (SW) piping, the Division 2 SW piping, and the CST system piping and pumps. All ruptures of the CST system were assumed to result in the loss of the condensate storage tanks as a water source for High Pressure Core Spray (HPCS) and RCIC.

The three flooding scenarios examined include postulated pipe ruptures to the Division 1 SW, the Division 2 SW, and the CST system in the CRD/COND room. The plant transient evolution was analyzed using these pipe ruptures as the event initiators, as depicted in the event trees shown in the Attachment.

The event trees are initially quantified assuming that flooding does not result in RCIC failure (i.e., the EDR drain line does not allow floodwater flow to spread from the CRD/COND pump room to the RCIC pump room). Results from these event tree analyses are provided in Table 1 as Cases 1, 2, and 3 (results are expressed in terms of CDF). The event trees are re-quantified

assuming an additional loss of RCIC due to flooding from the CRD/COND pump room through the unisolable EDR drain line. Results from these event tree analyses are provided in Table 1 as Cases 1R, 2R, and 3R. The analysis assumes if plant operators could not terminate flooding within one hour, then core damage would occur. The pipe rupture frequency was calculated based on the methodology of EGG-SSRE-9639, "Component External Leakage and Rupture Frequency Estimates," November 1991. The impact of the unisolable EDR drain line on CDF is obtained by summing the differences from each scenario [i.e., (Case 1R - Case 1) + (Case 2R - Case 2) + (Case 3R - Case 3)]. The event trees for each quantified and re-quantified scenario are provided following Table 1.

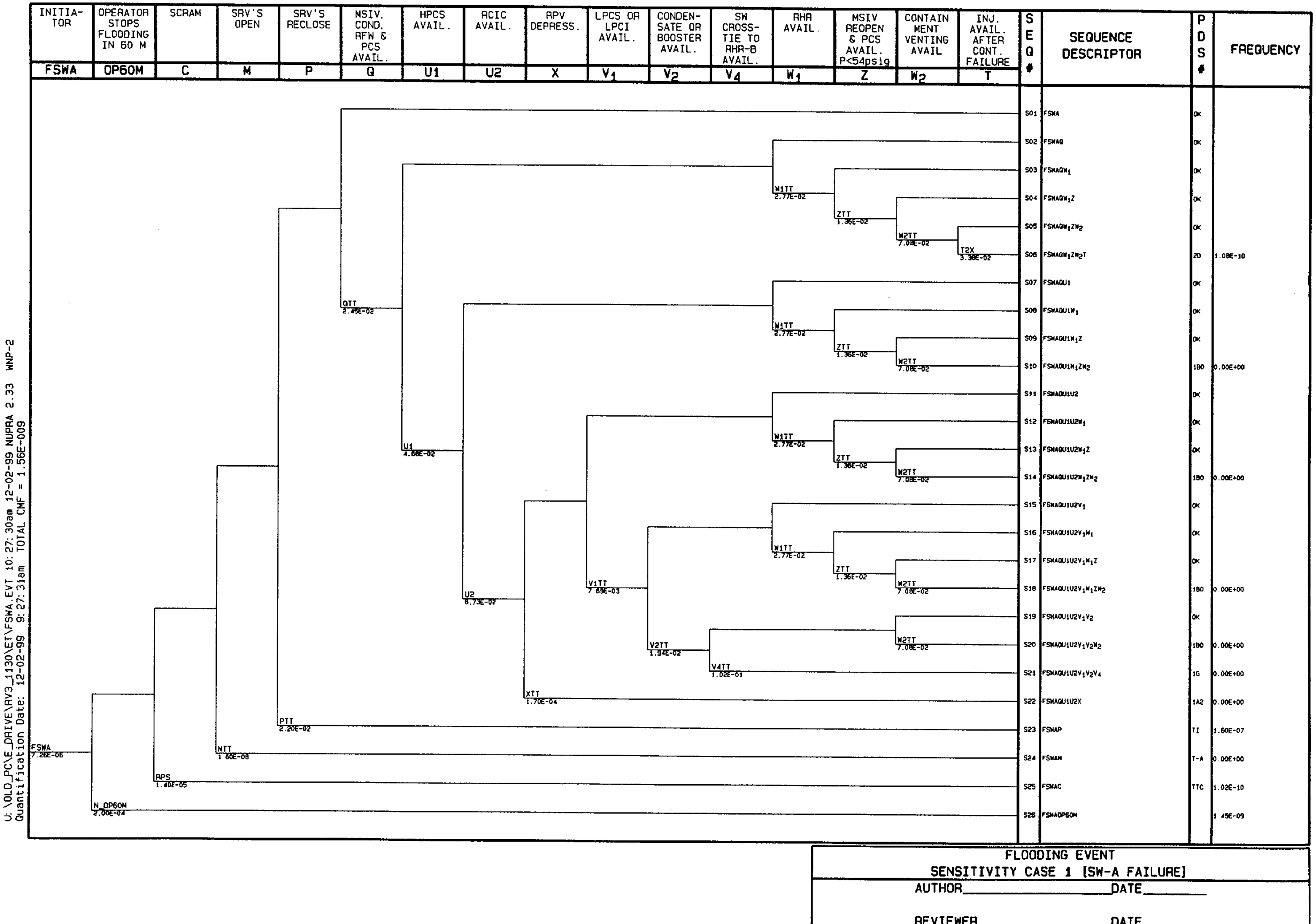
Conclusion

The results from the three scenarios are summarized in Table 1. The analysis uses a solution cutoff limit of 1E-10 (i.e., it includes CDF contributions from potential combinations of equipment failures that result in a CDF of 1E-10 or greater). The results show the impact of having an unisolable EDR drain line between the CRD/COND pump room and the RCIC pump room is less than 1E-10 per year increase to the CDF. Therefore it is concluded that the impact of the unisolable EDR drain line on CDF is negligible.

Table 1

PSA Impact of CRD Room Flooding

Transient Scenario Case	**Pipe Rupture Initiator**	**Rupture Frequency (per year)**	**System Out-of-Service**	**Core Damage Frequency (CDF)**	δ CDF [=CaseX−CaseXR]
1	SW-A	7.26E-6	SW-A	1.56E-9	Negligible
1R			SW-A, RCIC	1.56E-9	
2	SW-B	3.3E-6	SW-B	6.6E-10	Negligible
2R			SW-B, RCIC	6.6E-10	
3	Condensate Transfer (CST)	4.08E-4	CST	1.18E-10	Negligible
3R			CST, RCIC	1.18E-10	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Quantification Date: 12-02-99 9:27:31am TOTAL CMF = 1.56E-009
INITIATOR FSWA
OPERATOR STOPS FLOODING IN 60 M OP60M
SCRAM C
SRV'S OPEN M
SRV'S RECLOSE P
MSIV, COND, RFW & PCS AVAIL. Q
HPCS AVAIL. U1
RCIC AVAIL. U2
RPV DEPRESS. X
LPCS OR LPCI AVAIL. V1
CONDENSATE OR BOOSTER AVAIL. V2
SW CROSS-TIE TO RHR-B AVAIL. V4
RHR AVAIL. W1
MSIV REOPEN & PCS AVAIL. P<54psig Z
CONTAINMENT VENTING AVAIL W2
INJ. AVAIL. AFTER CONT. FAILURE T
SEQ #
SEQUENCE DESCRIPTOR
PDS #
FREQUENCY
FSWA 7.28E-06
N_OP60M 2.00E-04
RPS 1.40E-05
MTT 1.60E-08
PTT 2.20E-02
QTT 2.45E-02
U1 4.88E-02
U2 8.73E-02
XTT 1.70E-04
V1TT 7.69E-03
V2TT 1.94E-02
V4TT 1.02E-01
W1TT 2.77E-02
ZTT 1.36E-02
W2TT 7.08E-02
T2X 3.38E-02
S01 FSWA OK
S02 FSWAQ OK
S03 FSWAQW1 OK
S04 FSWAQW1Z OK
S05 FSWAQW1ZW2 OK
S06 FSWAQW1ZW2T 2D 1.08E-10
S07 FSWAQU1 OK
S08 FSWAQU1W1 OK
S09 FSWAQU1W1Z OK
S10 FSWAQU1W1ZW2 1B0 0.00E+00
S11 FSWAQU1U2 OK
S12 FSWAQU1U2W1 OK
S13 FSWAQU1U2W1Z OK
S14 FSWAQU1U2W1ZW2 1B0 0.00E+00
S15 FSWAQU1U2V1 OK
S16 FSWAQU1U2V1W1 OK
S17 FSWAQU1U2V1W1Z OK
S18 FSWAQU1U2V1W1ZW2 1B0 0.00E+00
S19 FSWAQU1U2V1V2 OK
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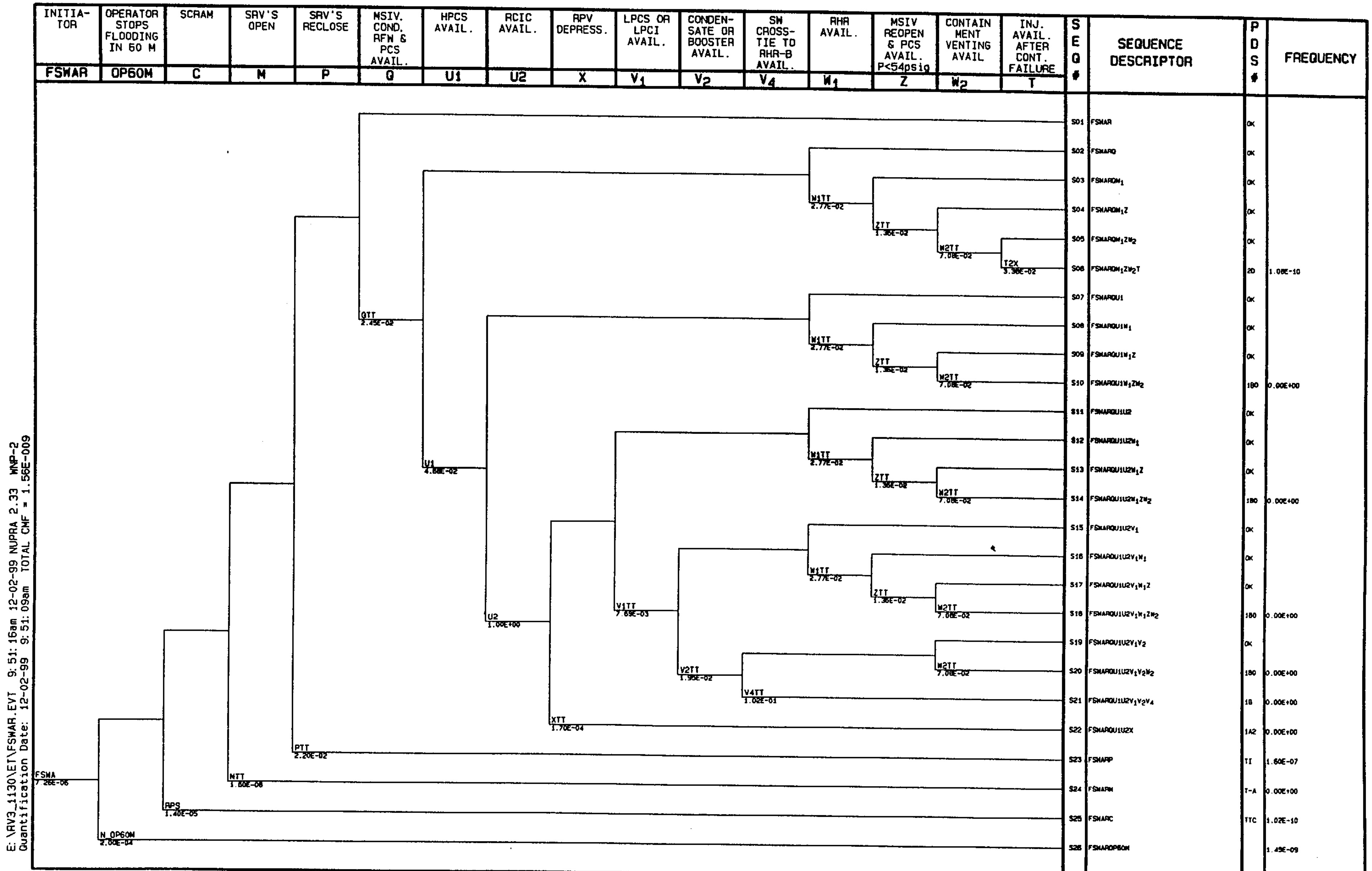
S21 FSWAQU1U2V1V2V4 1G 0.00E+00
S22 FSWAQU1U2X 1A2 0.00E+00
S23 FSWAP TI 1.60E-07
S24 FSWAM T-A 0.00E+00
S25 FSWAC TTC 1.02E-10
S26 FSWAOP60M 1.45E-09
FLOODING EVENT
SENSITIVITY CASE 1 [SW-A FAILURE]
AUTHOR
DATE
REVIEWER
DATE

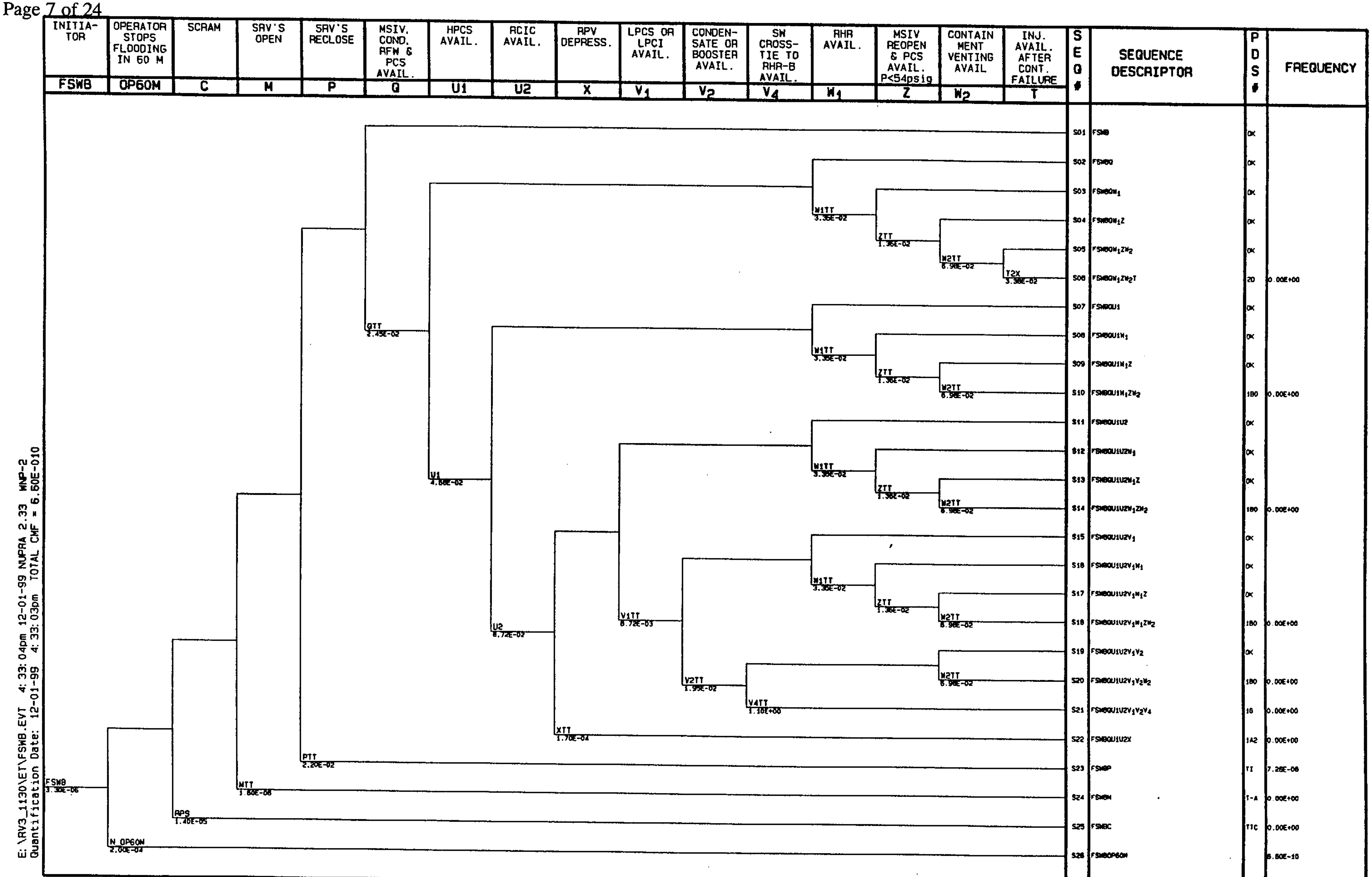


FLOODING EVENT
SENSITIVITY CASE 1R (SW-A, RCIC FAILURE)

AUTHOR________________ DATE________

REVIEWER________________ DATE________

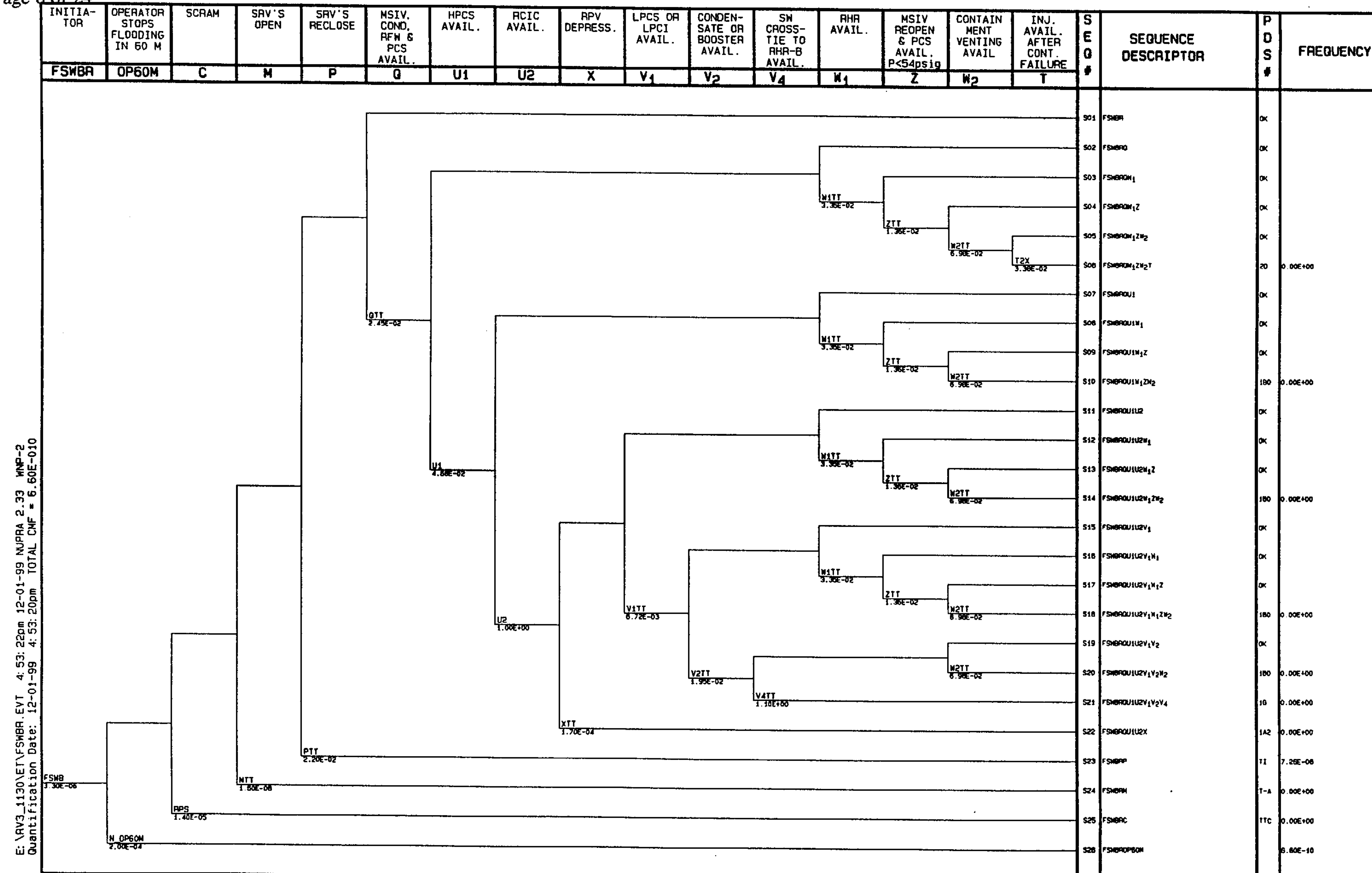
REQUEST FOR ADDITIONAL INFORMATION
ATTACHMENT



FLOODING EVENT
SENSITIVITY CASE 2 [SW-B FAILURE]

AUTHOR________________ DATE________

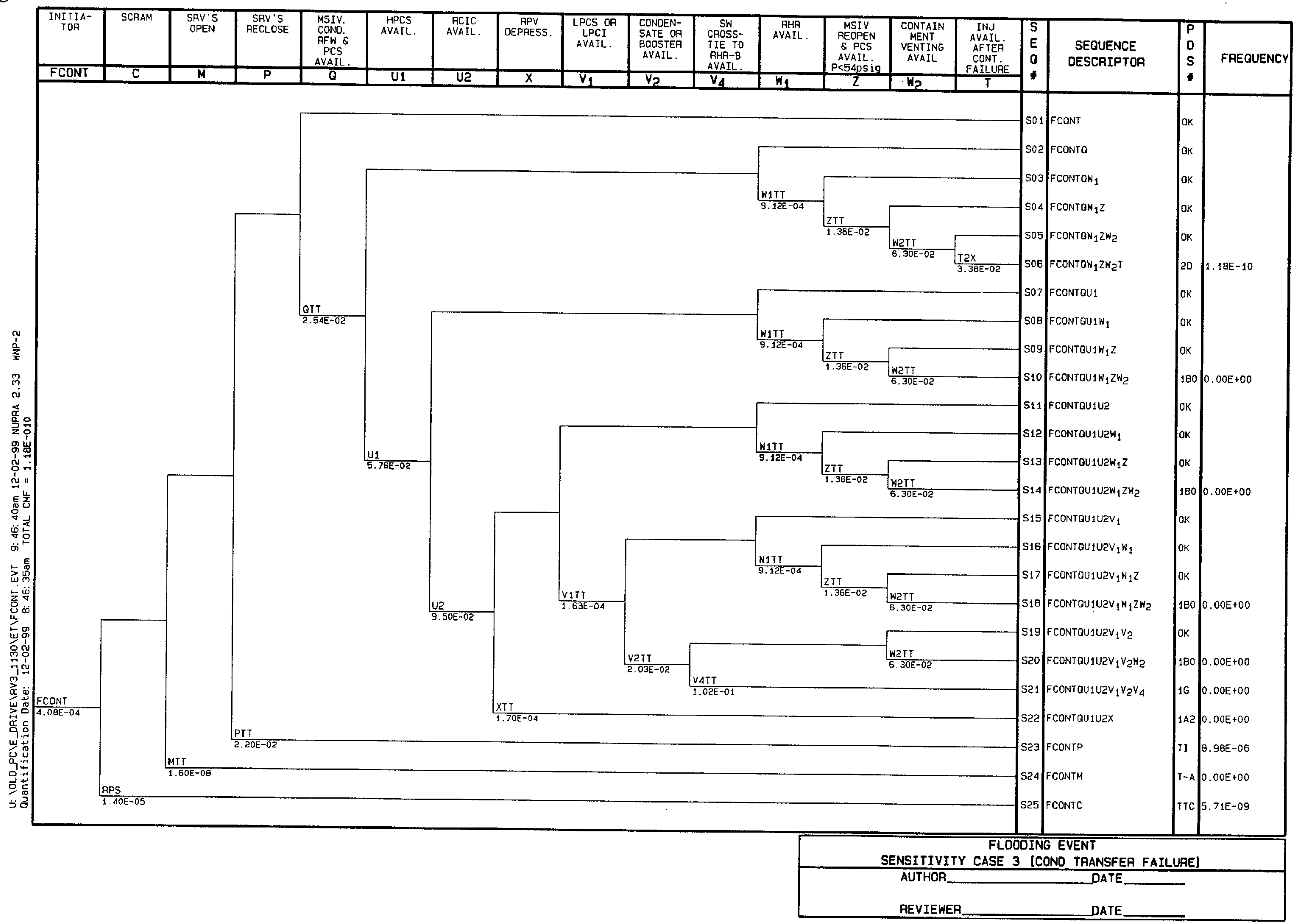
REVIEWER________________ DATE________



FLOODING EVENT
SENSITIVITY CASE 2A [SW-B, RCIC FAILURE]

AUTHOR__________ DATE______

REVIEWER__________ DATE______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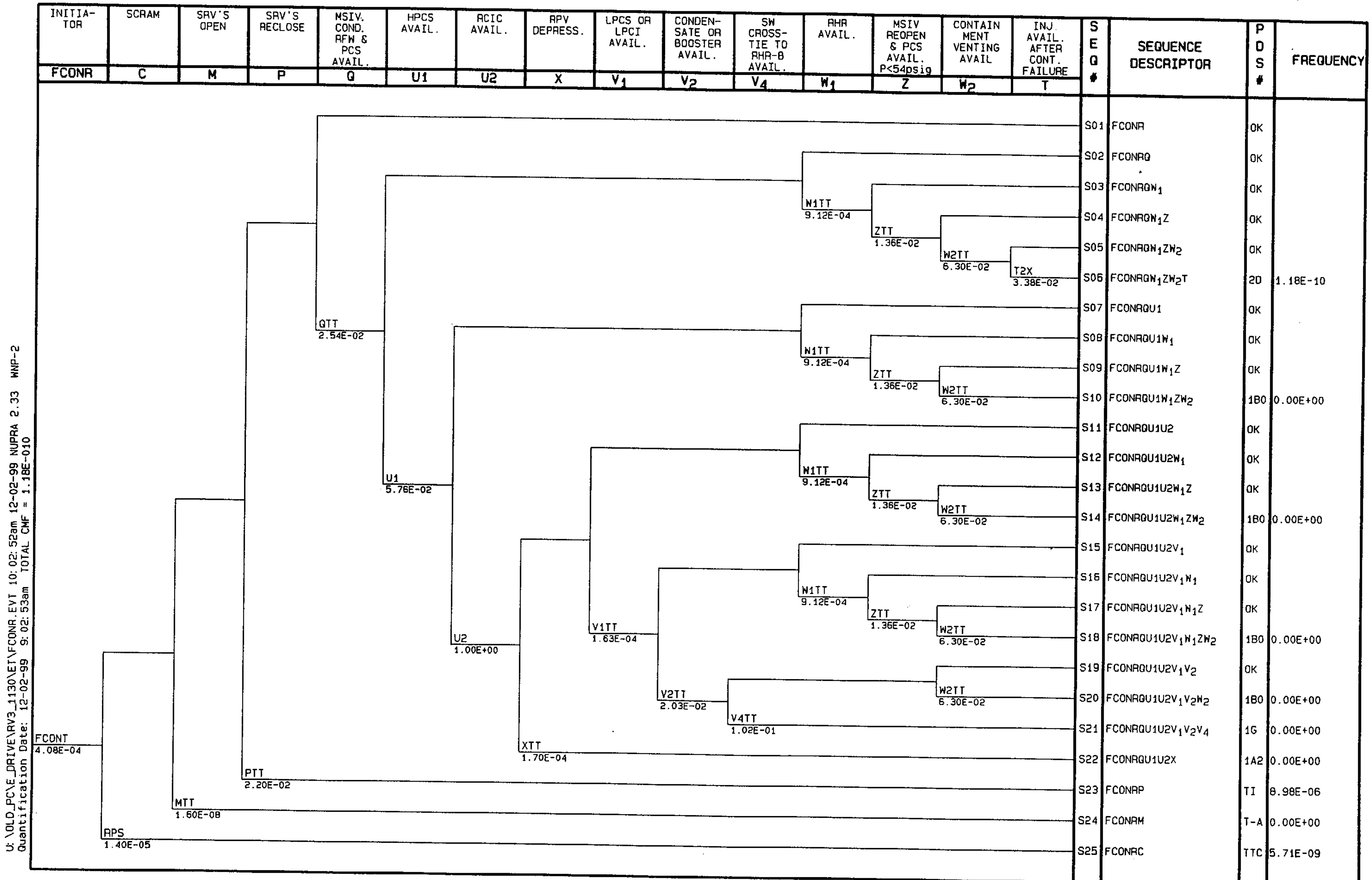
Quantification Date: 12-02-99 8:46:35am TOTAL CMF = 1.18E-010
INITIATOR FCONT
SCRAM C
SRV'S OPEN M
SRV'S RECLOSE P
MSIV. COND. RFW & PCS AVAIL. Q
HPCS AVAIL. U1
RCIC AVAIL. U2
RPV DEPRESS. X
LPCS OR LPCI AVAIL. V_1
CONDENSATE OR BOOSTER AVAIL. V_2
SW CROSS-TIE TO RHR-B AVAIL. V_4
RHR AVAIL. W_1
MSIV REOPEN & PCS AVAIL. P<54psig Z
CONTAINMENT VENTING AVAIL W_2
INJ. AVAIL. AFTER CONT. FAILURE T
SEQ #
SEQUENCE DESCRIPTOR
PDS #
FREQUENCY
FCONT 4.08E-04
RPS 1.40E-05
MTT 1.60E-08
PTT 2.20E-02
QTT 2.54E-02
U1 5.76E-02
U2 9.50E-02
XTT 1.70E-04
V1TT 1.63E-04
V2TT 2.03E-02
V4TT 1.02E-01
W1TT 9.12E-04
ZTT 1.36E-02
W2TT 6.30E-02
T2X 3.38E-02
S01 FCONT OK
S02 FCONTQ OK
S03 FCONTQW_1 OK
S04 FCONTQW_1Z OK
S05 FCONTQW_1ZW_2 OK
S06 FCONTQW_1ZW_2T 2D 1.18E-10
S07 FCONTQU1 OK
S08 FCONTQU1W_1 OK
S09 FCONTQU1W_1Z OK
S10 FCONTQU1W_1ZW_2 1B0 0.00E+00
S11 FCONTQU1U2 OK
S12 FCONTQU1U2W_1 OK
S13 FCONTQU1U2W_1Z OK
S14 FCONTQU1U2W_1ZW_2 1B0 0.00E+00
S15 FCONTQU1U2V_1 OK
S16 FCONTQU1U2V_1W_1 OK
S17 FCONTQU1U2V_1W_1Z OK
S18 FCONTQU1U2V_1W_1ZW_2 1B0 0.00E+00
S19 FCONTQU1U2V_1V_2 OK
S20 FCONTQU1U2$V_1V_2W_2$ 1B0 0.00E+00
S21 FCONTQU1U2$V_1V_2V_4$ 1G 0.00E+00
S22 FCONTQU1U2X 1A2 0.00E+00
S23 FCONTP TI 8.98E-06
S24 FCONTM T-A 0.00E+00
S25 FCONTC TTC 5.71E-09
FLOODING EVENT
SENSITIVITY CASE 3 [COND TRANSFER FAILURE]
AUTHOR ____ DATE ____
REVIEWER ____ DATE ____



FLOODING EVENT
SENSITIVITY CASE 3A [COND TRANSFER, RCIC FAILURE]

AUTHOR________________ DATE________

REVIEWER________________ DATE________

NRC Question #2:

Please provide a summary of the flooding analysis that confirms the viability of the proposed safe shutdown pathway and the affected equipment.

Energy Northwest Response:

The limiting flooding event and the associated safe shutdown analysis for the bounding moderate energy line through wall crack which floods both the Reactor Core Isolation Cooling (RCIC) pump room and the Control Rod Drive/Condensate (CRD/COND) pump room is summarized below.

The postulated crack is in an 18-inch Turbine Service Water (TSW) line located in the RCIC pump room with the crack flow rate calculated to be 363.70 gpm. Flooding in the RCIC pump room also directly floods the CRD/COND pump room since an unisolable drain line connects these rooms. This drain line carries a portion of the floodwater from the RCIC room to the CRD/COND pump room and thereby increases the floor area subject to flooding. This reduces the rate of water level rise in the two rooms, thus, allowing more time for operator action to terminate the event.

In addition to the direct connection between the RCIC pump room and the CRD/COND pump room, the RCIC pump room contains a Floor Drain Radioactive (FDR) line that is connected directly to the Residual Heat Removal (RHR) A pump room through automatic isolation valve FDR-V-607. Similarly, the floor drain located in the CRD/COND room is connected to a sump located in the High Pressure Core Spray (HPCS) pump room through a drain line containing automatic isolation valve FDR-V-608. Single failure of a drain line isolation valve to close results in additional flooding in the associated connected room.

Flooding from the TSW line pipe crack in the RCIC pump room potentially affects the equipment and cables located in this room as well as those in the CRD/COND pump room. This flooding also affects equipment served by cables passing through these rooms. Cables and equipment located in rooms that are adjacent to the RCIC/CRD pump rooms are also affected as a result of door leakage. The components located inside the pump rooms along with equipment connected to cables passing through these rooms that can potentially be affected by the TSW pipe crack are listed in Table 2 (CRD/COND Pump Room Affected Components) and Table 3 (RCIC Pump Room Affected Components).

Table 4 provides a summary of the flooding/safe shutdown analysis for the TSW pipe crack event. This summary includes the worst case assumed single failures, flood detection method, assumed event termination time, resultant water level flood height in adjacent or connected pump rooms, major equipment directly or indirectly affected by the flood that could be used for safe shutdown, and safe shutdown equipment that remains unaffected.

TABLE 2

CRD/COND Pump Room Components

Equipment Potentially Affected	
Cable No.	**Component Functional Description**
AM7CA-9080	Feeder, Sump Pump, FDR-P-2
AM7CA-9212	Power, 480V, Receptacle
AP3DA-9204	Transformer Feeder LP-30A-K
AP7AE-9060	Space Heater, CRD-P-1A
ASL53-9064	Motor Feeder, COND-P-4
ASM7-9090	Motor Feeder, CRD-P-1A
BM6BA-9070	Feeder, Pump COND-P-3
BM8CA-9020	Feeder, Pump FDR-P-4B
BM8CA-9030	Feeder, Pump, EDR-P-5
BP6BA-9204	Transfrmr Feeder, LP-6BA-J
BP8AE-9012	SPARE
BP8AE-9070	Space Heater, CRD-P-1B
BSL63-9051	Motor Feeder, COND-P-5
BSM2-9020	4160V Swgr SM-4 Normal Feeder
BSM8-9090	Motor Feeder Pump CRD-P-1B
3HPCS-10	Motor Feeder, MO Valve, HPCS-V-15
3HPCS-20	Motor Feeder, MO Valve, HPCS-V-1
3HPCS-30	Motor Feeder, MO Valve, HPCS-V-4
3HPCS-40	Motor Feeder, MO Valve, HPCS-V-12
3HPCS-50	Motor Feeder, MO Valve, HPCS-V-10
3HPCS-60	Motor Feeder, MO Valve, HPCS-V-11
3HPCS-80	Motor Feeder, HPCS water-leg pump P-3
3HPCS-170	Feeder, Valve SW-V-54
3HPCS-180	Feeder, RR-FN-4
3HPCS-340	Motor Feeder, HPCS Pump Motor
3HPCS-610	Htr Feeder, HPCS-P-1
AM7CA-9140	Feeder, RRA-FN-7
AM7CA-9150	Feeder, Htr RRA-EHC-7
AANN-9149	Alarm, COND-PS-45
AANN-9161	Alarm, COND-PS-43
AANN-9179	Alarm, RCC-FIS-6A
AANN-9180	Alarm, RCC-FIS-6B
AANN-9181	Alarm, RCC-FIS-6B1
ACOMA-9199	Data logging, HPCS-P-1
ACOMA-9203	Data logging, CRD-P-1A
ACOMA-9204	Data logging, CRD-P-1B
ACOMA-9205	Data logging, CRD-P-1A

TABLE 2

CRD/COND Pump Room Components

Equipment Potentially Affected	
Cable No.	**Component Functional Description**
ACOMA-9206	Data logging, CRD-P-1B
ACRD-9009	Ann, TB-R441 in CRD-IR-1B, TP"W-37"
ALDS-9001	Drywll Equip Dr Smp Flo, EDR-FT-37
ALDS-9002	Flow Transmitter, EDR-FT-37
ALDS-9003	RB Flr Dr Sump Flow, FDR-FT-38
AMISC-9232	Motion Rcdr, TR-1, Sensor SMA-1
AMISC-9233	Motion Rcdr, TR-2
AMISC-9236	Motion Rcdr, TR-1, 2, 3
AMISC-9237	Seismic Air Supply, SAP-1
AMISC-9238	Response Spectra Ann, RSP-1
AMISC-9239	Response Spectra Ann, RSP-1
AMISC-9240	Response Spectra Ann, RSP-1
AMISC-9585	Indication, RRA-TE-18
AM7CA-9084	Control, Lvl Sw, FDR-LS-6A
ASCAN-9022	Temp Recording, HPCS-P-1
ASCAN-9026	Temp Recording, CRD-P1A
ASCAN-9027	Temp Recording, CRD-P1B
ASL53-9072	Control, COND-P-4
ASL53-9074	Control, Press Sw 63, COND-PS-42
ASM7-9094	Control, CRD-PS-1A in CRD-IR-13
BANN-9133	Alarm, COND-PS-41
BANN-9134	Alarm, RCC-FIS-7
BANN-9180	Alarm, RCC-TS-6A1
BARM-9019	Rad Mon, Aux Unit Ch 13, Control Pnl
BCRD-9008	Ann, CRD-DPIS-15 in CRD-IR-1A
BCRD-9010	Ann, CRD-PS-1B in CRD-IR-1C
BM6BA-9071	Control, Sta, COND-P-3
BM6BA-9072	Control, PS-40, Condensate
BM8CA-9033	Control, Lvl Sw, EDR-LS-14A
BP8AE-9058	Htr Control for HPCS Pump Motor
BRWGE-9653	Control & Indication, Pos Sw, EDR-V-21
BRWGE-9657	Control, EDR-V-17, IR-61
BRWGE-9665	Control, FDR-LS-7B, IR-61
BRWGE-9667	Control & Indication, Pos Sw, FDR-V-7
BRWGE-9669	Control, FDR-V-603, IR-61
BRWGE-9673	Control, IR-61, FDR-AO-576B
BRWGE-9703	Control, Lvl Sw, FDR-P-4B, FDR-LS-5B
BRWGE-9707	Control, Lvl Sw, EDR-P-5, EDR-LS-14B

TABLE 2

CRD/COND Pump Room Components

Equipment Potentially Affected	
Cable No.	**Component Functional Description**
BRWGE-9751	Control, Lvl Sw, FDR-LS-1B
BRWGE-9838	Signal TE, EDR-TE-15
BSL63-9069	Control, COND-P-5
BSL63-9071	Control, COND-P-5
BSM8-9094	Control, Breaker CRD Pump 1B
1IR62-10	Control, COND-FIS-41, IR-62
1IR62-20	Control, COND-FIS-43, IR-62
1IR62-53	Control, COND-FCV-41, IR-62
1IR62-54	Control, COND-FCV-43, IR-62
1IR62-61	Control, COND-FSPV-41, IR-62
1IR62-62	Control, COND-FSPV-43, IR-62
2IR61-10	Control, COND-FIS-45, IR-61
2IR61-21	Control, COND-FCV-45, IR-61
2IR61-22	Control, COND-ESPV-45, IR-61
2NS4-4	Control FDR-V-3/EDR-V-19
2P8AE-14	Power, PP-8A-E, IR-61
3HPCS-11	Control, HPCS-V-15
3HPCS-12	Control, MCC HPCS MC-4A
3HPCS-15	Control for HPCS-V-15
3HPCS-17	Control for HPCS-V-15
3HPCS-21	Control, HPCS-V-1
3HPCS-22	Control, MCC HPCS MC-4A
3HPCS-23	Alarm, MCC HPCS MC-4A
3HPCS-31	Control, HPCS-V-4
3HPCS-32	Control, MCC HPCS MC-4A
3HPCS-41	Control, HPCS-V-12
3HPCS-42	Control, MCC HPCS MC-4A
3HPCS-51	Control, HPCS-V-10
3HPCS-52	Control, MCC HPCS MC-4A
3HPCS-61	Control, HPCS-V-11
3HPCS-62	Control, MCC HPCS MC-4A
3HPCS-71	Control, HPCS-V-23
3HPCS-72	Control, MCC HPCS MC-4A
3HPCS-82	Alarm, MCC HPCS MC-4A
3HPCS-91	Control for HPCS Pump Motor
3HPCS-92	Control for HPCS Pump Motor
3HPCS-94	Control, Supv Control Panel CS3
3HPCS-152	Control, MCC MC-4A

TABLE 2

CRD/COND Pump Room Components

Equipment Potentially Affected	
Cable No.	**Component Functional Description**
3HPCS-171	Control, Pos Sw SW-V-54
3HPCS-172	Control, MCC HPCS-MC-4A
3HPCS-182	Control Board, MCC HPCS MC-4A
3HPCS-191	Control, Supv Control Panel CS3
3HPCS-192	Control, MCC HPCS-MC-4A
3HPCS-201	Control, Supv Control Panel CS3
3HPCS-202	Control, Supv Control Panel CS3
3HPCS-210	MCC HPCS Remote Voltmeter
3HPCS-212	HPCS UV Alarm
3HPCS-213	HPCS Alarms, DG Aux Transformer
3HPCS-311	Breaker #4-2 Control, HPCS
3HPCS-314	4KV Swgr HPCS, Normal Source trip
3HPCS-321	4KV Swgr HPCS, Diesel Eng Control
3HPCS-322	4KV Swgr HPCS, Norm Supply Pt
3HPCS-323	4KV Swgr HPCS, Bus Pt Leads
3HPCS-324	4KV Swgr HPCS, UV Alarm
3HPCS-331	4KV Swgr HPCS, Brkr #4-DG3 Control
3HPCS-341	4KV Swgr HPCS, Brkr Control
3HPCS-343	4KV Swgr HPCS, Motor Feeder Trip & OL
3HPCS-344	4KV Swgr HPCS, HPCS Motor Amps
3HPCS-351	4KV Swgr HPCS, Brkr #4-41 Control
3HPCS-355	4KV Swgr HPCS, 480V Tran Trip Alarm
3HPCS-361	Gen Volt Reg Control
3HPCS-362	Diesel Engine Control
3HPCS-363	125VDC Panel Remote Voltmeter
3HPCS-364	Diesel Engine Governor Control
3HPCS-365	Diesel Engine Alarms
3HPCS-370	Protective Relay Panel
3HPCS-371	Diesel Engine Control
3HPCS-372	DG Point for Sync & Inst
3HPCS-373	HPCS Power Supply Alarms
3HPCS-374	DG Output Transducer
3HPCS-381	125VDC for Valve Pos Indication
3HPCS-382	HPCS system Relay Logic
3HPCS-403	HPCS AC Pwr Supply
3HPCS-404	Valve Control Power
3HPCS-421	120V AC Pwr, PP-4A
3HPCS-422	Signal, Supv Control Pnl

TABLE 2

CRD/COND Pump Room Components

Equipment Potentially Affected	
Cable No.	**Component Functional Description**
3HPCS-456	HPCS-V-5, Control Power
3HPCS-510	Signal, Supv Control Pnl
3HPCS-611	Control, Htr Contractor for HPCS-P-1
BARM-9030	Rad Mon, Aux Unit 23, HPCS Cabinet
BRWGE-9671	SPARE
BARM-9101	Audio Alarm CH13, TB-R353
BARM-9102	Audio Alarm CH13, TB-R353
BARM-9143	Audio Alarm CH23, TB-R353
BARM-9144	Audio Alarm CH23, TB-R353
BRWGE-9651	Control, EDR-AO-22, EDR-V-21, IR-61
BRWGE-9655	Control & Indication, Pos Sw, EDR-AO-22, TR-R14
BRWGE-9659	Control & Indication, Pos Sw, EDR-AO-17, TP-R14
BRWGE-9675	Control, Pos Sw FDR-AO-576B, TP-R14
BRWGE-9705	Control, EDR-P-5A, Lvl Sw, EDR-LS-14A

TABLE 3

RCIC Pump Room Components

Equipment Potentially Affected	
Cable No.	**Component Functional Description**
AM7CA-9211	Power, 480V receptacle
AM7CA-9213	Power, 480V receptacle
AM7CA-9214	Power, 480V receptacle
AM7CA-9215	Power, 480V receptacle
AM7CA-9219	Power, 480V receptacle
BM8CA-9010	Feeder, Pump, FDR-P-1B
BM8CA-9040	Feeder, Pump, FPC-P-3
1M1ID-30	Feeder, Valve, RCIC-V-10
1M1ID-170	Feeder, Valve, RCIC-V-46
1M1ID-10	Feeder, Valve, RCIC-V-31
1M2IA-130	Feeder, Valve, RCIC-V-19
1M2IA-140	Feeder, Valve, RCIC-V-22
1M2IA-150	Feeder, Valve, RCIC-V-59
1M2IA-200	Feeder, Valve, RCIC-V-69
1M2IA-10	Feeder, T/G Trip Throttle Vlv w/ RCIC-DT-1
1M2IA-170	Feeder, Valve, RCIC-V-45
1M2IA-180	Feeder, Valve, RCIC-P-2
1M2IA-190	Feeder, Valve, RCIC-P-4
1SM7-50	Feeder LPCS Pump Motor
1M7B-340	Feeder RCIC Pump 3 Motor
2M8B-450	Feeder, RRA-FN-6
2SM8-60	Feeder RHR Pump C Motor
ARCIC-9065	Alarm, RCIC-PIS-34
ARCIC-9064	Alarm, Div 1 Supervisory Control Panel
ARCIC-9063	Alarm, RCIC-P-2
ARCIC-9066	Alarm, RCIC-LS-10
ARCIC-9061	Alarm, RCIC-LSL-12
ARCIC-9062	Alarm, RCIC-LSH-1L
ARCIC-9052	Control, Pos Sw & Sol, SW-V-34
AP7AE-9062	Space Htr, RCIC-P-2, 120VAC
AP7AE-9063	Space Htr, RCIC-P-4, 120VAC
AIVD-9108	Indication, Pos Sw RCIC-V-19
AIVD-9110	Indication, Pos Sw RCIC-V-31
AIVD-9106	Indication, Pos Sw RCIC-V-69
BANN-9223	Alarm, SW-FS-29
BANN-9224	Alarm, SW-TS-29
BRCIC-9022	Alarm, RCIC-LS-4

TABLE 3

RCIC Pump Room Components

Equipment Potentially Affected	
Cable No.	**Component Functional Description**
BARM-9053	Audio Alarm, Aux Unit CH 12
BARM-9054	Audio Alarm, CH 12
BARM-17	Rad Mon, Aux Unit CH12
BARM-9018	Rad Mon, Aux Unit CH12
BMISC-9525	Control, RCIC-SPV-19A
1LDS-104	Temp Recorder, LD-TE-25A
1LDS-110	Diff Temp Recorder, LD-TE-5A
1LDS-111	Temp Sw, LD-TE-4A
1LDS-109	Diff Temp Sw, LD-TE-26A
1RCIC-52	Deleted, Chg to ARCIC-9065
1RCIC-46	Alarm, Rad Bldg, RCIC-FIC-1R
1RCIC-58	Alarm, Turbine RCIC-DT-1
1RCIC-8	Indication, Turbine, RCIC-DT-1
1RCIC-27	Control, Turbine, RCIC-DT-1
1RCIC-31	Control, Rad Bldg, RCIC-DT-1
1RCIC-32	SPARE
1RCIC-63	Control, Turbine RCIC-DT-1
1RCIC-40	SPARE
1RCIC-5	Control, Steam Ln, Pot, Lvl Sw, RCIC-LS-10
1RCIC-24	Control, Pos Sw, RCIC-V-45
1RCIC-34	Control, Pos Sw Valve, RCIC-V-45
1RCIC-41	SPARE
1RCIC-2	Indication, Limit Sw on RCIC-V-26
1RCIC-48	Deleted, Chg to ARCIC-9061
1RCIC-4	Indication, Limit Sw on RCIC-V-5
1RCIC-29	Vacuum Tank, RCIC-LSH-11
1RCIC-47	Alarm, Vac Tnk RCIC-PSH-13
1RCIC-28	Position Mon on RCIC-V-31
1P7AE-13	Power, PP-7A-E
1M2IA-11	Control, Pos Sw on Turb Trip Valve
1M21A-12	Control, overspeed LS on Turb RCIC-DT-1
1M21A-15	Control, overspeed LS on Turb RCIC-DT-1
1M2IA-122	SPARE
1M2IA-171	Control, Pos Sw on RCIC-V-45
1M2IA-208	Control, Pos Sw on RCIC-V-45
1M2IA-207	Control, Pos Sw on RCIC-V-45
1M2IA-141	Control, Pos Sw, RCIC-V-22
1M2IA-151	Control, Pos Sw, RCIC-V-59

TABLE 3

RCIC Pump Room Components

Equipment Potentially Affected	
Cable No.	**Component Functional Description**
1M2IA-131	Control, Pos Sw, RCIC-V-19
1M2IA-201	Control, Pos Sw, RCIC-V-69
1M1ID-31	Control, Pos Sw, RCIC-V-10
1M1ID-11	Control, Pos Sw, RCIC-V-31
2LDS-48	Temp Sw, TE LD-TE-24B
2LDS-47	Temp Sw, TE LD-TE-25B
2LDS-53	Temp Sw, TE LD-TE-5B
2LDS-52	Temp Sw, TE LD-TE-26B
2LDS-54	Temp Sw, TE LD-TE-4B
2LDS-63	Temp Sw, TE LD-TE-6B
2M8B-452	Control, MC-8B
2M8B-451	Control, RRA-FN-6
2RCIC-2	Indication, RCIC-V-25
2RCIC-4	Indication, RCIC-V-4
2RCIC-16	Pos Mon, RCIC-V-45
2SM8-93	Control Brkr SW1B
1RCIC-15	SPARE
ALDS-9110	Diff Temp Sw, LD-TE-5A
1M1ID-161	Control, Pos Sw RCIC-V-46
1M2IA-101	SPARE
1RCIC-7	Lvl Indication for RCIC-V-54
1RCIC-49	Deleted, Chg to ARCIC-9062
1RCIC-50	Deleted, Chg to ARCIC-9063
1RCIC-51	Deleted, Chg to ARCIC-9064
1RCIC-53	Deleted, Chg to ARCIC-9066
ALDS-9111	Temp Sw, LD-TE-4A

TABLE 4

Safe Shutdown Analysis Summary

Single Failure	**Flood Detection Method**	**Safe Shutdown Analysis Summary**
RCIC Leak Detection	RHR-C, RHR-B and HPCS room level switches or operator walkdown at max interval of 12 hours.	Flooding is assumed to be detected first by HPCS pump room level switch (4 gpm from door leakage plus 5 gpm from drainage into HPCS Floor Drain-Radioactive [FDR] Sump) in 341 minutes. Event termination, 30 minutes after detection, results in a flood height in the HPCS pump room at 6.5 inches; much less than the 69 inches (i.e., the Emergency Operating Procedure [EOP]) safe level for this room. The flood heights in adjacent RHR-B and RHR-C pump rooms due to through-door leakage will be lower. The EOP safe flood heights for these rooms are 72 inches for the RHR B pump room and 67 inches for RHR C pump room. The event is terminated by operator action as described in the response to NRC Question 3. If no door leakage is assumed in the HPCS pump room, a maximum of 12 hours is assumed for flood detection. Twelve hours of undetected flooding into RCIC/CRD pump rooms results in a flood level of 10.4 feet in these rooms. Although cabling passing through these rooms for the HPCS, Low Pressure Core Spray (LPCS) and RHR C pump motors is located above the 10.4 foot flood level, these circuits are conservatively assumed to be lost along with all equipment/cabling in the rooms. This results in the loss of RCIC, HPCS, LPCS, and RHR C systems. The minimum set of safe shutdown equipment unaffected by the flood in conjunction with the worst case single failure is: Reactor Protection System (RPS) (SCRAM), Main Steam Isolation Valves (MSIVs) (reactor vessel isolation), Automatic Depressurization System/Safety Relief Valve (ADS/SRV) (reactor vessel depressurization), RHR B (alternate shutdown cooling mode), and supporting Service Water, HVAC, Division 2 Diesel Generator, and the Division 2 DC system.

TABLE 4

Safe Shutdown Analysis Summary

Single Failure	Flood Detection Method	Safe Shutdown Analysis Summary
RHR-A or RHR-B	RCIC room level switch	The flow rate into rooms RCIC/CRD includes 25 gpm assumed drainage into the EDR sump. Event termination, 30 minutes after detection, results in a flood level of 11.2 inches in the RCIC/CRD rooms. This is above the EOP safe flood level of 6 inches for the RCIC room. RCIC is assumed lost. The flood heights in the HPCS, RHR-B and RHR-C pump rooms due to door leakage at flood termination are less than 11.2 inches in each room. This level is below the EOP safe flood levels for these rooms which is 69 inches for the HPCS pump room, 72 inches for the RHR B pump room, and 67 inches for the RHR C pump room. The event is terminated by operator action as described in the response to NRC Question 3. The minimum set of safe shutdown equipment unaffected by the flood in conjunction with the worst case single failure is: RPS (SCRAM), MSIVs (reactor vessel isolation), ADS/SRV (reactor vessel depressurization), RHR A or B depending on single failure (alternate shutdown cooling mode), and supporting Service Water, HVAC, Diesel Generator, and the DC battery systems.

TABLE 4

Safe Shutdown Analysis Summary

Single Failure	Flood Detection Method	Safe Shutdown Analysis Summary
FDR-V-608	RCIC or HPCS room level switches or RHR-B, RHR-C, or LPCS room level switches if door leakage occurs	The net flow rate into rooms RCIC/CRD and HPCS (through the floor drain cross-tie line) is 363.70 gpm, from the TSW crack in the RCIC room, plus 25 gpm drainage into the EDR sump, plus 5 gpm drainage into the HPCS FDR sump. Event termination, 30 minutes after detection, results in a flood height in the RCIC/CRD and HPCS pump rooms at 10.3 inches. This is greater than the minimum EOP safe level of 6 inches for the RCIC room. RCIC is assumed lost. This height is much lower than the safe EOP flood height in the HPCS pump room which is 69 inches. The event is terminated by operator action as described in the response to NRC Question 3. If 4 gpm door leakage occurs into adjacent rooms (RHR B, RHR C, and LPCS pump rooms), the flood heights in these pump rooms at flood termination will be less than 10.3 inches. The EOP safe flood levels are 72 inches for the RHR B pump room, 67 inches for RHR C, and 58 inches for the LPCS pump rooms. The minimum set of safe shutdown equipment unaffected by the flood in conjunction with the worst case single failure is: RPS (SCRAM), MSIVs (reactor vessel isolation), ADS/SRV (reactor vessel depressurization), RHR A and B (alternate shutdown cooling mode), and supporting Service Water, HVAC, Diesel Generator, and the DC battery systems.

TABLE 4

Safe Shutdown Analysis Summary

Single Failure	Flood Detection Method	Safe Shutdown Analysis Summary
FDR-V-607	RCIC or RHR A room level switches or HPCS, RHR-B, or RHR-C room level switches if door leakage occurs	The net flow rate into the pump rooms, RCIC/CRD and RHR-A (through the connecting floor drain crosstie line), is 363.70 gpm, plus 25 gpm drainage into the EDR sump. Event termination, 30 minutes after detection, results in flood heights in RCIC/CRD and RHR-A pump rooms of 10 inches which is greater than 6 inches (i.e., the EOP safe level for RCIC pump room). RCIC is assumed lost. However, this flood height is much less than the 36 inch EOP safe flood level for the RHR A pump room. The event is terminated by operator action as described in the response to RAI Question 3. If 4 gpm door leakage occurs into each adjacent room, the flood heights in the HPCS, RHR-B, and RHR-C pump rooms at flood termination will be lower than 10 inches which is below the EOP safe flood levels of 69 inches for the HPCS pump room, 72 inches for the RHR B pump room, and 67 inches for the RHR C pump room. The minimum set of safe shutdown equipment unaffected by the flood in conjunction with the worst case single failure is: RPS (SCRAM), MSIVs (reactor vessel isolation), ADS/SRV (reactor vessel depressurization), RHR A and B (alternate shutdown cooling mode), and supporting Service Water, HVAC, Diesel Generator, and the DC battery systems.

NRC Question #3

Part of the basis for the proposed change is the execution of timely operator action in response to a flood detected by safety-related leak detection sensing instrumentation in the reactor core isolation coolant pump room, thus limiting the amount of equipment potentially lost from the event. Please provide a human factors evaluation in accordance with ANSI/ANS 58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operations Actions," to estimate the response time required to terminate the flood.

Energy Northwest Response:

The human factors evaluation in accordance with ANSI/ANS 58.8-1984, for estimating the response time required to terminate the flood is summarized below.

The bounding moderate energy pipe crack associated with the Reactor Core Isolation Cooling/Control Rod Drive (RCIC/CRD) pump room connection is an 18-inch Turbine Service Water (TSW) line that is postulated to crack in the RCIC pump room. The mitigating action for this break is to trip the associated pump, either TSW-P-1A or 1B from the main control room using the pump control switch. This action will be executed per the Abnormal Flooding Procedure "ABN-FLOOD".

The estimated response time required for an operator to terminate the above flooding event is 26 minutes based on the guidance provided in ANSI/ANS 58.8-1984. This estimate includes 20 minutes (from Table 1 of ANSI/ANS 58.8-1984, assuming Plant Condition 5) for the operator to determine the source of the flood (after detection) and identify the appropriate mitigating action(s) required to terminate the event. In addition, the estimate includes an additional 6 minutes from Table 2 of ANSI/ANS 58.8-1984 (5 minutes + 1 minute per operator action, assuming Plant Condition 5) necessary for the operator to accomplish tripping of the pump by turning the pump motor control switch to the stop position. The time assumed in the Columbia Generating Station accident analysis for termination of the TSW line crack event is 30 minutes after detection. ANSI/ANS 58.8-1984 notes that the time periods specified in the criteria are considered a minimum time that can be credited for operator actions. Therefore, the assumed accident analysis event duration of 30 minutes is conservative with respect to ANSI/ANS 58.8-1984 guidance. The resulting water level in the flooded areas for the 30-minute event termination time will be higher and the potential for equipment loss greater than for the 26-minute event termination time.