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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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September 12, 1996
GO2-96-181

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO INDIVIDUAL PLANT EXAMINATION FOR WASHINGTON PUBLIC POWER SUPPLY SYSTEM (WPPSS) NUCLEAR PROJECT NO. 2 (WNP-2)

References:

1) Letter, GI2-96-207, dated August 12, 1996, TG Colburn (NRC) to JV Parrish (SS), "Request for Additional Information - Washington Public Power Supply System (WPPSS) - Washington Project No. 2 (WNP-2) (TAC No. M74489)"

2) Letter, GI2-95-179, dated August 7, 1995, JW Clifford (NRC) to JV Parrish (SS), "Request for Additional Information Related to Individual Plant Examination for Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2) (TAC No. M74489)"

3) Letter, G02-94-175, dated July 27, 1994, JV Parrish (SS) to NRC, "Revision 1 to Response to Generic Letter 88-20, 'Individual Plant Examination for Severe Accident Vulnerabilities 10 CFR 50.54(F)'"

4) Letter, G02-95-224, dated October 20, 1995, JV Parrish (SS) to NRC, "Response to Request For Additional Information Related To Individual Plant Examination For Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2)"

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO INDIVIDUAL PLANT EXAMINATION FOR WNP-2

This letter responds to the Reference 1 request for additional information (RAI). The request resulted from the staff's review of the Supply System's Individual Plant Examination (IPE) submittal (Reference 3) and responses to the original RAI (Reference 2). Appendix A contains a restatement of each question and the corresponding response. The responses (Reference 4) to the original RAI that required requantification of the IPE models included an indication of probable outcome, where appropriate, and a determination that this would be incorporated in future revisions to the models. In the second RAI, the Staff has expressed concern over the use of qualitative terms and the completeness of the IPE in terms of the initiating events analyzed. The quantification shows the additional support system initiating events do not contribute to core damage frequency and the qualitative discussion provided in our original response was accurate. The Staff was concerned about the understanding of the interface between core cooling and containment cooling capabilities particularly with regard to the high pressure injection systems. The requantification confirms the conservatism in the original submittal and shows, even with the correction to HPCS/RCIC use, the long term loss of containment cooling sequence frequencies decrease from 2.12E-06/Rx-yr to 1.86E-06/Rx-yr. The use of fire protection water as a mitigating system was questioned and the response herein corrects the text inconsistencies noted in our original response and reports the results of the use of fire protection water for Station Blackout scenarios. The requantification is complete and the information contained in Appendix A is based on the requantification. The responses show quantitatively that the use of the qualitative terms used in the original response (Reference 4) was appropriate and the conclusions insensitive to the changes requested. As noted in Reference 4, the Supply System recognizes the usefulness of the models and techniques resulting from the IPE and is incorporating them into PSA format for certification when that process is defined.

Reference 3 and 4, as supplemented with the information in the attachment to this letter, represents a complete response to the original request for information contained in Generic Letter 88-20. Therefore, the Supply System does not intend to submit a revised WNP-2 IPE for the docket.

Should you have any questions or desire additional information regarding this matter, please call me or Mr. R. E. Levline at (509) 377-4549.

Respectfully,

D. A. Swank
Manager, Regulatory Affairs
Mail Drop PE20

Attachment

cc: LJ Callan - NRC RIV
KE Perkins, Jr. - NRC RIV, WCFO
NS Reynolds - Winston & Strawn
TG Colburn - NRR
DL Williams - BPA 399
NRC Sr. Resident Inspector - 927N

1. Initiating Events

The licensee did not model in the IPE the loss of certain support systems that may cause a reactor trip as initiating events. Events such as loss of heating, ventilation, and air conditioning (HVAC) to the emergency bus electrical switchgear rooms, loss of Division 1 direct current (DC) electrical power, loss of individual alternating current (AC) buses, and loss of reactor building component cooling water were apparently not specifically modeled or quantitatively evaluated as initiating events. It is stated in the licensee's response to the staff's request for additional information (RAI) that loss of these systems have "negligibly small" frequencies or their occurrence was assumed to be adequately accounted for through the evaluation of the loss of other support systems. Although an initiating event frequency of 1E-4 may be small, its contribution to core damage frequency (CDF) may not be negligible if it results in a CDF of 1E-6. In addition, loss of these support systems has the potential of affecting the availability or operability of many systems simultaneously and the systems affected can differ depending on the support system. Therefore, grouping the loss of one support system under another may not be appropriate.

The staff is concerned that the licensee did not perform an in-depth, systematic analysis of plant specific initiating events, particularly with regard to the loss of the support systems mentioned above. Please provide (a) the exact values for those initiators (e.g., loss of all HVAC and loss of Division 1 of DC power) for which you concluded that their contribution to the CDF is negligibly small and the bases for these values; and (b) the bases (e.g., failure mode and effects analyses) for concluding that loss of all DC power is not a credible event, and that loss of AC buses is enveloped by loss of DC buses due to battery depletion.

RESPONSE:

The initiating events considered for the WNP-2 IPE, Revision 2 (hereinafter identified as the WNP-2 PSA) have been expanded, particularly for support systems that may be initiators. Table Q1-1 compares the IPE report list (see Table 4-3 in the IPE, Rev. 1) with the current PSA list. The current list is based on a failure modes and effects analysis of all WNP-2 systems and the values derived from LER data from start of commercial operation through December of 1995. As can be determined from the list, loss of various AC system buses and loss of DC Division 1 have been added. The loss of Standby Service Water is not included since loss of its function does not cause a reactor scram and, therefore, does not meet the definition of an initiating event. The reactor building component cooling water (RCCW) provides cooling water to potentially contaminated heat exchangers in the reactor building. The failure modes and effects analysis identified a loss of cooling to the primary containment coolers as resulting in high reactor building pressure SCRAM. RPS actuations are included in the turbine trip initiating event frequency. The RCCW supports CRD injection which was not credited due to its small flow rate. Therefore, RCCW as an initiator is included only as a contributor to turbine trip.

In the WNP-2 PSA model, HVAC/room cooler failures are modelled as contributors to ESF system failures except for the RCIC system. For RCIC, engineering calculation shows that the RCIC room does not overheat and continuous pump operation is anticipated with or without the room coolers. The contribution of room cooler failure to the system unavailability is as follows,

HPCS	4.7%
LPCS	16.2%
LPCI-A	28.0%
LPCI-B	24.0%
LPCI-C	19.2%

In addition, basic events that involve HVAC/room cooler failures are also included for the Standby Service Water pump rooms, the EDG rooms and the critical switchgear rooms. In these system analyses, loss of room cooling directly causes the functional failure of these systems. The loss of HVAC to ECCS equipment rooms was, therefore, included in the sequences as a failure mode, but was not included as an initiating event as it would not cause a plant scram or ESF actuation.

Loss of a single electrical switchgear room will not directly cause an initiating event, but will result in minimizing mitigation capability. To be considered as an initiating event in the control room or critical switchgear rooms, the loss of HVAC would need to occur, backup room cooling would need to fail, the room would have to heat up beyond equipment service temperatures and failure of equipment that causes plant trip or safety system actuation would need to occur before repair is completed. This assertion is verified by a test of area temperature upon taking out of service an HVAC that supplies one division of safety related batteries, battery chargers, inverters, 4160 switchgear, and 480 volt switchgear. In five hours, the maximum room temperature increase recorded was approximately 4F. However, in response to the question, the Supply System has performed analyses to investigate the effects of loss of HVAC in the control room or the critical switchgear room based on the work done in NUREG/CR-6084. The analyses are briefly summarized as follows;

<u>Loss of Control Room HVAC:</u> The initiating event frequency of 4.49E-02/Rx-yr was assumed and the probability that repair fails prior to reaching high temperatures in the control room is taken as 4.3E-3 (see NUREG/CR-6084). The operator action on high control room temperature is proceduralized and although no credit was taken for recovery actions, it was assumed the plant would enter shutdown prior to loss of equipment due to high temperature. The resultant contribution to the CDF, as shown on the event tree, is negligible.

<u>Loss of Critical Switchgear Room HVAC:</u> The initiating event frequency and probability of non-repair in NUREG/CR-6084 were used. The loss of HVAC flow and room temperature are annunciated in the control room with procedures for response. Therefore, a 0.1 factor was used

for non-recovery probability assuming knowledge-based human actions. The critical divisional power in the switchgear room was assumed to be lost. As seen from the attached event tree, the contribution to CDF is negligible.

The specific responses to the requested information are:

a) Table Q1-2 is a comparison of IPE initiating events and the current WNP-2 initiating events contribution to Core Damage Frequency (CDF). As seen in the table, the additional initiating events of Loss of Div 1 DC contributed 1.98E-08/yr which is less than Loss of Div 2 DC as stated in the original response; however, the loss of HVAC, loss of RCCW, and loss of an AC bus did not contribute at all to the CDF.

b) The bases for concluding that loss of all DC is not a credible initiating event is i) there have been no complete loss of DC power events at WNP-2, ii) a survey of other BWRs failed to identify any that considered complete loss of DC as an initiating event, and iii) the loss of single bus DC initiating event frequency includes a conservative 50% recovery factor, whereas, realistically the complete loss of DC initiator is estimated at 4E-07/Rx-yr which would result in a CDF contribution approximately equal to the loss of a single DC bus. The loss of individual AC buses have been included in Table Q2-1 rather than qualitatively bounding it by a loss of DC bus due to battery depletion.

With the above information, the WNP-2 PSA includes a comprehensive initiators list based on industry knowledge and a plant specific failure analysis of WNP-2 systems, including the support systems, for potential initiators.

TABLE Q1-1

Initiating Event Frequencies

	Initiation Event	Rev 1 Frequency Events/Year	Rev 2 Frequency Events/Year
1	General Transients		
	• Turbine Trip	3.30	2.90
	• MSIV Closure	0.2	0.115
	• Loss of Condenser	0.05	0.06
	• Loss of Feedwater	0.1	0.5
	• Loss of Offsite Power	2.46×10^{-2}	2.46×10^{-2}
	• IORV/SORV	0.2[a]	0.19[a]
	• Manual Shutdown	0.5	1.4
2	LOCA		
	• Large LOCA (D > 6″)	3×10^{-4}	3×10^{-4}
	• Medium LOCA (4″ < D < 6″)	3×10^{-3}	3×10^{-3}
	• Small LOCA (1″ < D < 4″)	8×10^{-3}	8×10^{-3}
	• Steam Line Break Outside Containment	2.17×10^{-5}	2.17×10^{-4}
	• ISLOCA	1.21×10^{-6}	2.26×10^{-3}
3	ATWS[c]		
	• ATWS following Turbine Trip with Bypass (100% power)	2.7	2.4
	ATWS following Turbine Trip with Bypass (25% power)	0.6	0.5
	• ATWS following MSIV Closure	0.2	0.115
	• ATWS following Loss of Condenser	0.05	0.06
	• ATWS following Loss of Feedwater	0.1	0.5
	• ATWS following SORV	0.2	0.19
4	Special Initiators		
	• Loss of a Division 2 DC	3×10^{-3}	3×10^{-3}
	• Loss of a Division 1 DC	not included	3×10^{-3}
	• Loss of TSW	1.25×10^{-3}	1.25×10^{-3}
	• Loss of CIA	included as part of CN	1.25×10^{-3}
	• Loss of an AC Bus	not included	$2\text{-}4 \times 10^{-4}$
	• Loss of HVAC		
	-Control Room	not included	1.93×10^{-4}
	-Switchgear Rooms	not included	8.39×10^{-6}

•	Loss of CN	1.25×10^{-3}	included as part of CIA
•	Instrument Line Break	1×10^{-2}	1×10^{-2}
•	Internal Flooding (Category 6)	2.92×10^{-3}	2.92×10^{-3}
•	Internal Flooding (Category 7)	1.6×10^{-5}	2.70×10^{-7}
•	Internal Flooding (Category 14)	4.69×10^{-3}	4.69×10^{-3}
•	Loss of Standby Service Water	1.83×10^{-4}	not included
•	Loss of CAS	1.25×10^{-3}	1.25×10^{-3}

NOTES

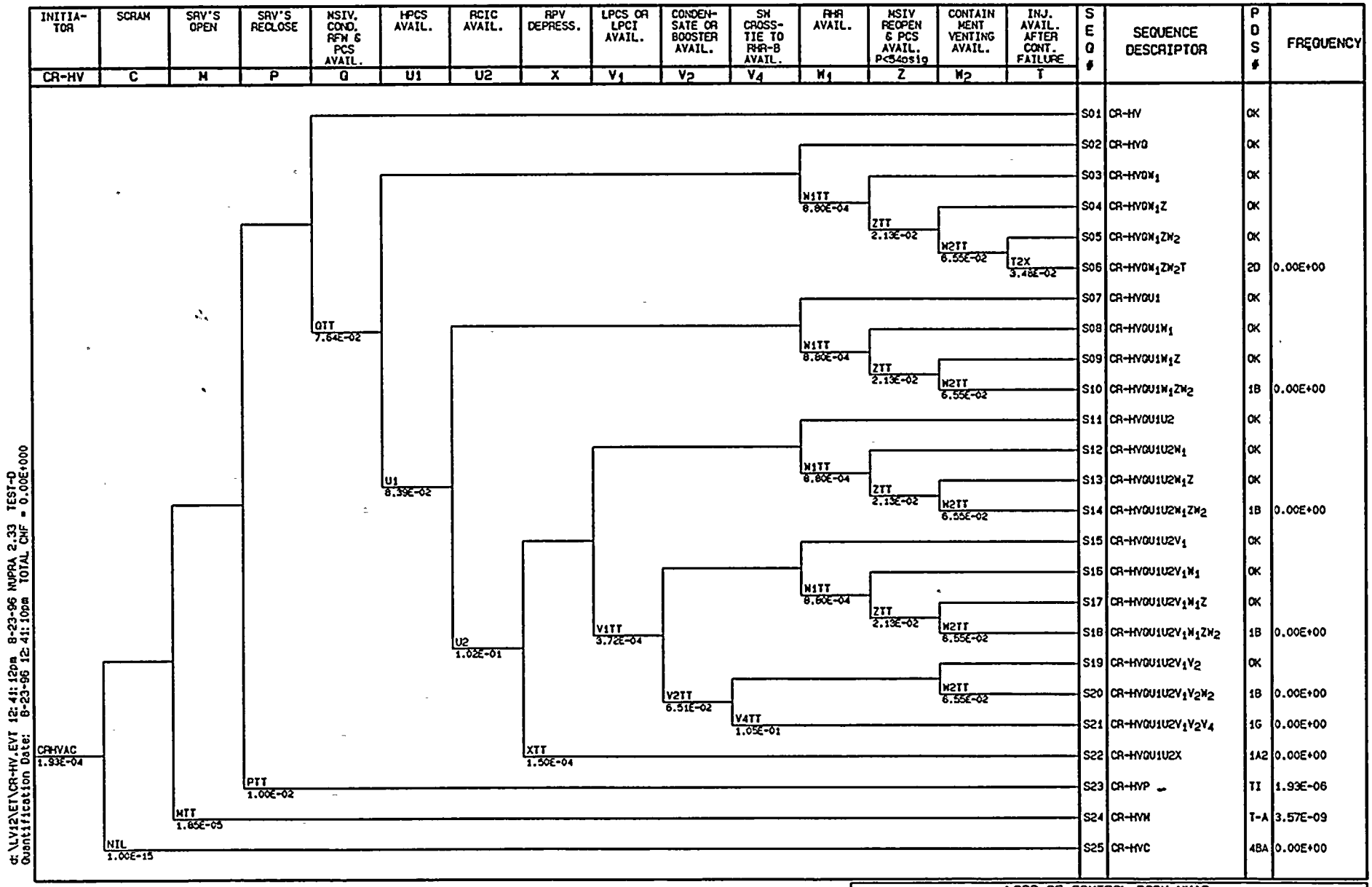
a Frequency includes transfers from other event trees.

c The ATWS event trees use these transient frequencies as initiating events and are followed in the trees by events for failures of the mechanical and electrical portions of RPS.

TABLE Q1-2

	WNP-2 IPE REV. 1 TOTAL CDF = 1.75E-005 /yr		WNP-2 IPE REV. 2 TOTAL CDF = 1.43E-005 /yr	
INITIATORS	frequency per year	% of CDF	frequency per year	% of CDF
T(E): LOSS OF OFFSITE POWER	1.18E-005	67.2%	1.04E-005	73.1%
FLD7: FLOODING CASE 7	1.68E-006	9.6%	2.64E-007	1.8%
TSSW: LOSS OF STANDBY SERVICE WATER	6.21E-007	3.6%	not included	
FLD14: FLOODING CASE 14	5.22E-007	3.0%	3.69E-007	2.6%
TT: TURBINE TRIP	4.78E-007	2.7%	5.17E-007	3.6%
TTC: TURBINE TRIP ATWS (100% POWER)	4.37E-007	2.5%	1.40E-007	1.0%
FLD6: FLOODING CASE 6	3.16E-007	1.8%	2.12E-007	1.5%
TC: LOSS OF CONDENSER	3.07E-007	1.8%	3.04E-007	2.1%
SR: INSTRUMENT LINE BREAK	2.70E-007	1.5%	1.18E-007	0.8%
TDC2: LOSS OF DIV2 DC	2.13E-007	1.2%	7.18E-008	0.5%
TDC1: LOSS OF DIV1 DC	not included		1.98E-008	0.1%
TTSW: LOSS OF PLANT SERVICE WATER	1.69E-007	1.0%	2.57E-007	1.8%
AO: LARGE LOCA OUTSIDE CONTAINMENT	1.50E-007	0.9%	5.96E-009	0.0%
TCAS: LOSS OF CONTROL AND SERVICE AIR	1.30E-007	0.7%	6.55E-008	0.5%
TMC: MSIV CLOSURE ATWS	1.05E-007	0.6%	2.26E-008	0.2%

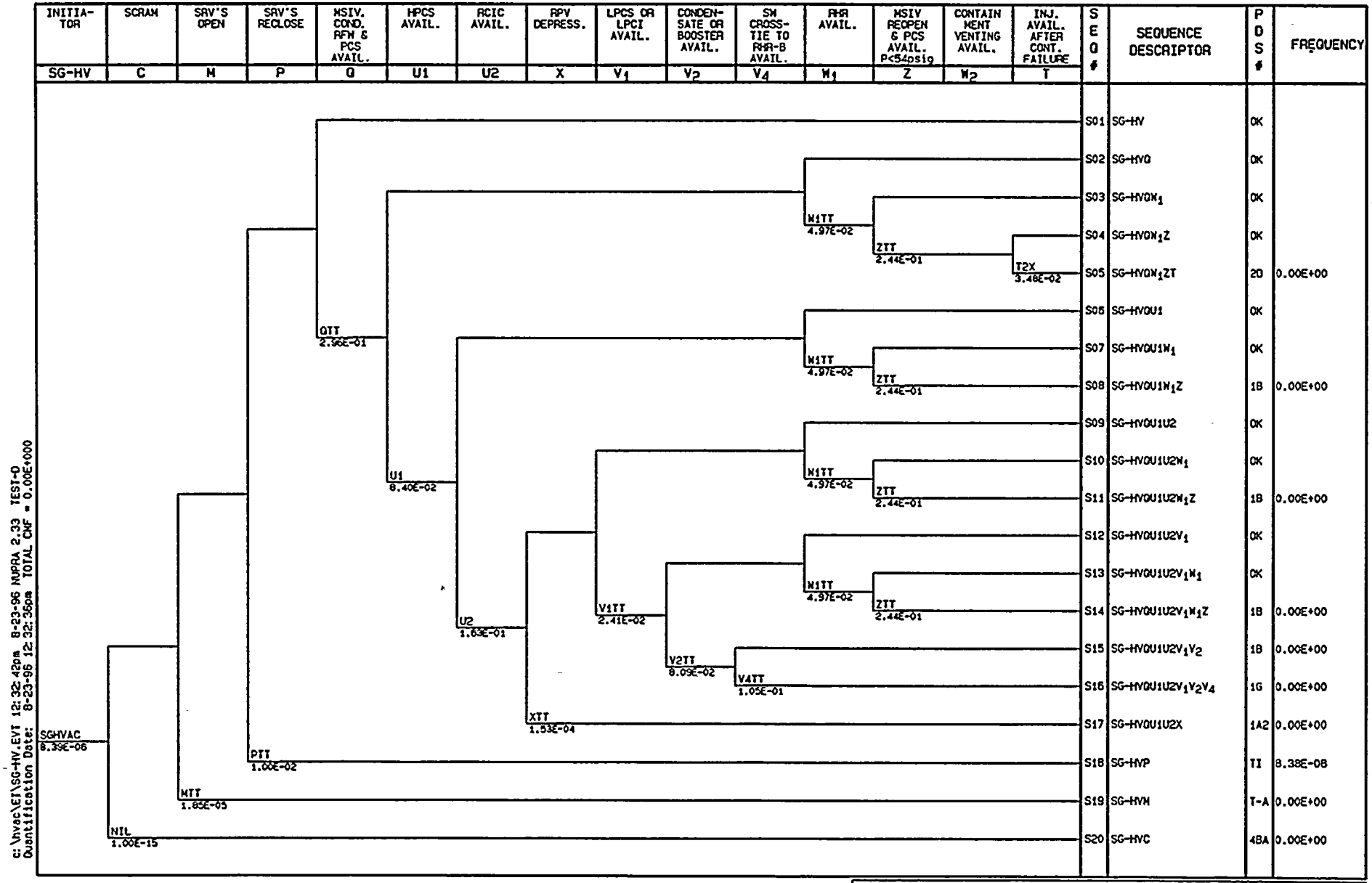
MS: MANUAL SHUTDOWN	5.41E-008	0.3%	2.20E-007	1.5%
TFC: LOSS OF FEEDWATER ATWS	5.25E-008	0.3%	1.12E-007	0.8%
S1: MEDIUM LOCA	5.10E-008	0.3%	8.21E-009	0.1%
TCN: LOSS OF CN TIA: LOSS OF CIA	TCN 4.69E-008	0.3%	TIA 2.52E-009	0.0%
TF: LOSS OF FEEDWATER	3.69E-008	0.2%	8.39E-007	5.9%
A: LARGE LOCA	3.45E-008	0.2%	3.90E-009	0.0%
TCC: LOSS OF CONDENSER ATWS	2.51E-008	0.1%	1.12E-008	0.1%
TM: MSIV CLOSURE	1.81E-008	0.1%	7.60E-008	0.5%
TI: IORV/SORV	7.69E-009	0.0%	1.02E-007	0.7%
TIC: SORV ATWS	5.78E-009	0.0%	1.04E-007	0.7%
S2: SMALL LOCA	1.44E-010	0.0%	0.00E+000	0.0%
TTC2: TURBINE TRIP ATWS (25% POWER)	0.00E+000	0.0%	0.00E+000	0.0%
IS: ISLOCA	0.00E+000	0.0%	0.00E+000	0.0%
LOSS OF AC BUS: TSM3 TSM2 TSH6 TSH5 TSM1	not included		0.00E+000	0.0%
LOSS OF HVAC: CONTROL ROOM SWITCHGEAR ROOM	not included		0.00E+000	0.0%



LOSS OF CONTROL ROOM HVAC
FIGURE 3.1.2.1-2

AUTHOR_____DATE_____

REVIEWER_____DATE_____



LOSS OF ESF SWITCHGEAR HVAC
FIGURE 3.1.2.1-28

AUTHOR_____DATE_____

REVIEWER_____DATE_____

2. Interface Between Core Cooling and Containment Heat Removal

The IPE submittal and the RAI responses indicate that the licensee may not have comprehensively evaluated the impact of loss of containment cooling on the ability to provide both short and long term cooling of the core. For example, the IPE assumed that reactor core isolation cooling (RCIC) and high pressure core spray (HPCS) systems would be able to provide core cooling even under conditions where the containment has failed due to overpressurization and suppression pool temperatures exceeding 212 F. Under such conditions, RCIC is expected to trip due to high turbine exhaust pressure and HPCS may fail due to excessive temperatures causing bearing and seal failure and/or due to loss of adequate net positive suction head (NPSH). In addition, with loss of containment cooling, the containment repressurizes and ultimately the safety relief valves (SRVs) may be forced closed. Upon closure of the SRVs, the reactor pressure vessel will repressurize and lose low pressure injection. These phenomena do not appear to have been taken into account. The staff is concerned that the licensee did not examine the interfaces between containment conditions and core cooling systems thoroughly and comprehensively. The information provided in the submittal and response do not indicate that a systematic and comprehensive assessment of each system's capability to perform satisfactorily under adverse conditions such as loss of containment cooling or containment failure was made. The licensee indicated during a conference call with the staff (April 29, 1996) that they are in the process of addressing these limitations of the IPE. Please provide the following:

a. The revised core damage frequency.

b. A description of the new important sequences and their contributors.

c. A description of the accident progression for each sequence involved in the General Transient Event Tree. The description should include the assumptions and system success criteria used for both the OK and CD branches in relation to different containment conditions (containment cooled, containment not cooled, containment failed).

RESPONSE:

The interface between core cooling and containment cooling (or containment failure) has been clarified. The loss of containment cooling will cause loss of RCIC on high backpressure prior to containment failure. Therefore, if core cooling is solely dependent on RCIC, the injection fails prior to containment failure and core damage results. If the HPCS system is successful, automatic switchover of the injection pump suction to the suppression pool from the CST would occur whenever the pool reached a predetermined level. With no containment heat removal, containment pressure will eventually reach the "Maximum Primary Containment Water Level Limit" (MPCWLL). The EOPs instruct the operators to terminate injection into the primary containment from sources external to the primary containment. Note that HPCS injection from

the CST is a viable way to maintain vessel makeup for an additional 21 hours and this allows opportunities for recovery of RHR, PCS or vent. However, because of EOP instruction, no credit is taken to include the operator action of establishing CST as the suction source.

For cases with HPCS available, continued injection from the pool is assured up to containment failure (HPCS pump failure temperature is 360 F due to seal, gasket, etc. beginning to fail which corresponds to 150 psia saturated pressure. Adequate NPSH is maintained up to containment failure). Containment failure from severe overpressure can take two forms:

- catastrophic failure resulting in a large breach, rapid depressurization and possibly bulk boiling of the suppression pool. This causes loss of all injection and core melt because:
  - there is insufficient available NPSH for the pumps upon containment depressurization so they cavitate and fail,
  - release of steam to the reactor building at a location which causes high temperatures in the ECCS pump rooms and resultant failure of pump motors or switchgear.
- a self limiting membrane tear which results in a controlled leak to the reactor building. This in turn may release enough steam at the right location in the reactor building to cause the temperature in the pump rooms to increase enough to initiate failure of the injection pump motors or critical switchgear. This eventually leads to core melt. A membrane tear is always expected to occur before catastrophic failure if the containment pressure increase is gradual.

Based on WNP-2 containment structural analyses, containment failure has been demonstrated to preferentially occur in any one of three locations. Two are located in the drywell region, and one is located above the horizontal stiffeners in the wetwell. The conditional occurrence probability for each failure location was found to be equal, so there is a 33% chance that the failure will occur in the wetwell and a 67% chance that it will occur in the drywell. When the containment breach is located in the wetwell region, it is assumed that the ECCS pumps would fail.

In the event that the ECCS pumps fail after containment failure, the operator could still depressurize the vessel and establish vessel makeup with an external source, e.g., Standby Service Water cross-connected to RHR-B.

The impact of accounting for loss of RCIC and the potential loss of HPCS on the long term containment cooling failure sequences (typically designated TW sequences) has been evaluated and has been shown to be:

SUMMED FREQUENCIES OF TW SEQUENCES	
Revision 1 of IPE Report	2.12E-06/Rx-yr
Revision 2	1.86E-06/Rx-yr

Therefore, as stated in our response to the original RAI, the realistic modeling assumptions result in decreasing the combined frequencies of the TW sequences to the total CDF. Specifically, in response to the requested information:

a. The Core Damage Frequency has been revised to provide more realistic and complete analysis of WNP-2 and has been reduced from 1.75E-05/Rx-yr to 1.43E-05/Rx-yr in the current revision. This value reflects the additional initiating events mentioned in the RAI, as well as credit for the Fire Protection system mentioned in the following question.

b. The quantification of the accident sequences results in a core damage frequency of 1.43E-5/Rx-year from all internal initiating events and internal flooding. The dominant contributor to core damage is the station blackout scenarios which account for approximately 72 percent of the total core damage frequency. Table Q2-1 lists the 53 sequences with percent contribution to the total core damage frequency greater than 1%.

TABLE Q2-1
Summary of Accident Sequence Quantification Results

Sequence	Frequency	% of TCDF (=1.43E-5)	Sequence Name
T(E)S18	5.21E-006	36.5%	T(E)SMU(1)REC
T(E)S16	4.00E-006	28.0%	T(E)SMREC
T(E)S22	6.39E-007	4.5%	T(E)SMU(1)U(2)VREC
TTS06	4.81E-007	3.4%	TTQW(1)ZW(2)T
TFS17	4.49E-007	3.1%	TFU1U2X

TFS16	3.28E-007	2.3%	TFU1U2V(1)V(4)
FLD14S03	2.88E-007	2.0%	FLD14W(1)T
TTSWS03	2.57E-007	1.8%	TTSWW(1)T
FLD7S01	2.56E-007	1.8%	FLD7
TCS04	2.28E-007	1.6%	TCW(1)W(2)T
T(E)S26	2.09E-007	1.5%	T(E)SMU(1)PREC
MSS06	2.04E-007	1.4%	MSQW(1)ZW(2)T
T(E)S24	1.96E-007	1.4%	T(E)SMU(1)U(2)XREC
FLD6S03	1.72E-007	1.2%	FLD6W(1)T
T(E)S06	1.29E-007	0.9%	T(E)U(1)W(1)REC
TTCS18	1.23E-007	0.9%	TTCC(M)C(3)
SRS17	1.18E-007	0.8%	SRUX
TFCS30	1.01E-007	0.7%	TFCC(M)C(3)
TICS07	9.30E-008	0.7%	TICCC(3)
TIS21	9.22E-008	0.6%	TIU1U2X
FLD14S05	8.10E-008	0.6%	FLD14U1W(1)

TDC2S06	7.18E-008	0.5%	TDC2QW(1)ZW(2)T
TMS21	7.07E-008	0.5%	TMU1U2X
TCASS03	6.55E-008	0.5%	TCASW(1)T
TCS07	6.16E-008	0.4%	TCU1W(1)W(2)
TFS05	5.69E-008	0.4%	TFW(1)ZW(2)T
T(E)S11	4.42E-008	0.3%	T(E)U(1)U(2)VREC
FLD6S05	3.99E-008	0.3%	FLD6U1W(1)
TTS10	3.64E-008	0.3%	TTQU1W(1)ZW(2)
TMCS30	2.26E-008	0.2%	TMCC(M)C(3)
Remaining Sequences Contribute Less Than 0.1% Each			

Table Q2-1 shows that the only initiating events whose sequences sum to greater than 3% of the total core damage frequency are:

- Loss of Offsite Power (TE)
- Turbine Trip (TT)
- Loss of Feedwater (TF)

The event trees for TE, TT and TF are attached.

Of the events listed in Table Q2-1, five sequences contribute greater than 3% to the CDF. The top three are identical to the top three contributors in the IPE submittal. The TSW flood that was fourth in the IPE submittal was reduced by performing more realistic analysis on its initiating frequency and the other top event, loss of offsite power with a diesel available, was reduced by the more realistic assumptions discussed in the response to Question 2b. of the original RAI. These five are listed below and discussed in the following sections:

WNP-2 IPE DOMINANT SEQUENCES (>3% contribution to CDF)

BRIEF SEQUENCE DESCRIPTION	FREQUENCY	% OF CDF
Station Blackout with HPCS failure and failure to recover offsite power in four hours	5.21E-06	36.5
Station Blackout with HPCS operating but failure to recover offsite power in ten hours	4.00E-06	28.0
Station Blackout with HPCS, RCIC, Fire Protection Water failure and failure to recover offsite power in thirty minutes	6.39E-07	4.5
Turbine Trip with HPCS operating and failure of long term decay heat removal	4.81E-07	3.4
Loss of Feedwater with loss of HPCS, RCIC and depressurization	4.49E-07	3.1

Station Blackout Lasting Greater Than Four Hours (Sequence TE-S18)

For the WNP-2 IPE/PSA, Station Blackout (SBO) scenarios are defined as having a loss of offsite AC power with coincident failures of Emergency Diesel Generators EDG-1 and EDG-2 which results in loss of AC power to 4Kv buses for Division 1 and Division 2.

Sequence TE-S18 results from loss of offsite AC power, station blackout and an additional failure of either the HPCS system or the HPCS diesel generator EDG-3. RCIC system injects successfully until the batteries become depleted or containment pressure reaches approximately 20 psig. All balance of plant and containment heat removal systems remain unavailable until offsite power is restored. If offsite power is not restored before RCIC is rendered unavailable by battery depletion or high containment pressure, the resulting loss of injection will initiate core uncovery and consequential core damage.

Station Blackout Lasting Greater Than Ten Hours (Sequence TE-S16)

Sequence TE-S16 represents a long term station blackout sequence, i.e., loss of offsite power followed by coincident failures of DG 1 and DG 2. High pressure core spray (HPCS) is available to provide injection since its operation is not limited by four hour battery lifetime. Successful HPCS injection for ten hours without containment decay heat removal results in CST inventory depletion occurring at this time, if that is the source used by the operator. Therefore, it is assumed HPCS fails, initiate core uncovery and core damage. To arrest the sequence before core damage, offsite power must be recovered before the CST inventory is depleted, about ten hours after the initiating event.

Station Blackout Without Injection (Sequence TE-S22)

Sequence TE-S22 represents a station blackout sequence which differs from TE-S18 only in the fact that RCIC is unsuccessful for mechanical reasons, and fire protection water is unsuccessful because of human error and hardware failures. Unless offsite power is recovered within 30 minutes of the failure of RCIC, the total loss of injection will result in core damage.

Turbine Trip with HPCS operating and failure of long term decay heat removal (Sequence TT-S06)

Sequence TT-S06 is initiated by a turbine trip event. Reactor SCRAM is successful and HPCS is available for injection initially. However, since decay heat removal is unavailable via RHR, the PCS or containment vent, containment temperature will rise due to suppression pool heat-up and eventually reach the failure point. In this sequence, HPCS fails after containment failure and the operating staff fails to depressurize the vessel and establish vessel makeup with Standby Service Water cross-connected to RHR-B. The total loss of injection will result in core damage.

Loss of Feedwater with loss of HPCS, RCIC and depressurization (Sequence TF-S17)

This is a loss of feedwater scenario in which HPCS and RCIC are unsuccessful for mechanical reasons, and the operating staff fails to depressurize the vessel. Since the RCS pressure remains high, the low pressure injection systems will not initiate and results in core damage.

c. The General Transient Event Tree is typified by the Turbine Trip Event Tree attached. The sequences are discussed in order from sequence S01 through S22.

S01: This represents expected response to an unplanned turbine trip from power. The scram functions to reduce power to decay heat levels, SRVs operate (if necessary), and core decay heat is removed via the condenser with vessel makeup from the Condensate/Feedwater system. Since core decay heat is being removed external to the primary containment, the condition of the containment systems is not questioned.

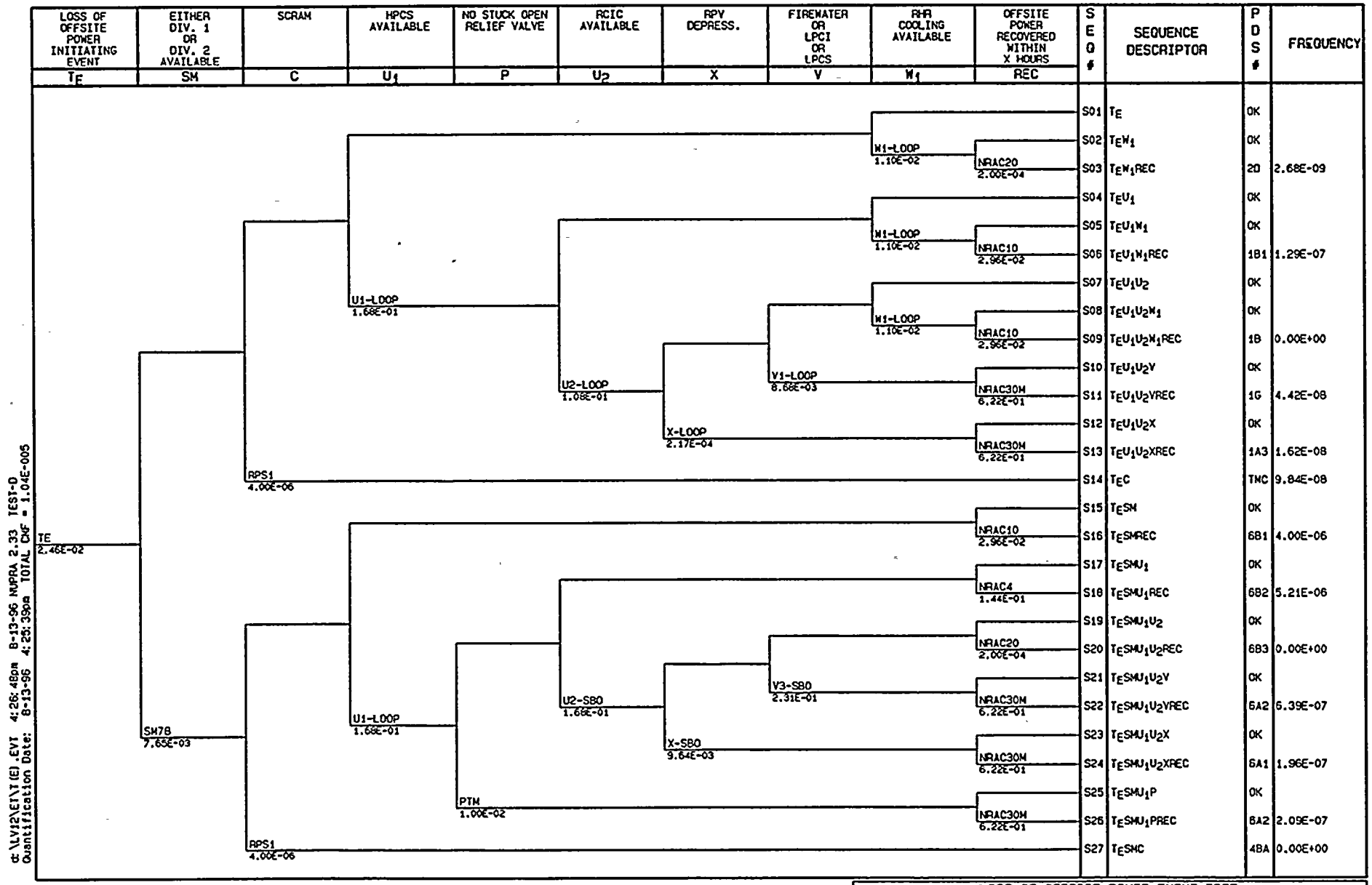
S02-S06: In this set of sequences, the event starts as before, however, the normal heat removal system of condenser/feedwater becomes unavailable. For sequences S02-S06, vessel inventory is being maintained via the high pressure core spray (HPCS). Core decay heat is being transferred to the suppression pool via SRVs. Sequence S02 end state is OK because RHR is available in suppression pool cooling mode and its system success criteria is such that containment cooling does not adversely impact HPCS operation, i.e., the success criteria is that RHR in suppression pool cooling mode is used to maintain suppression pool temperature below 212F per the FSAR. If RHR in suppression pool cooling mode fails, sequence S03 still ends in OK if the normal heat removal path of condenser/feedwater can be re-established before the containment pressure reaches 54 psig (pressure where MSIV reclose). The success criteria for this action is based on the fact that HPCS qualification temperature is not reached at this time. If the normal heat removal system (condenser) cannot be re-established before 54 psig is reached, then the containment heat removal path through the vent system is attempted. If successful, the core decay heat is being removed via HPCS to the suppression pool and containment heat is being removed via the vent system and Sequence S04 ends with OK. If the containment vent fails to function, then containment fails from overpressurization. Depending on the containment failure location, there is a possibility that the containment break does not impact HPCS operation and the break acts the same as the vent in removing containment heat. This is Sequence S05 which ends in OK. If HPCS and RHR/SW Crosstie are not available or failed after containment failure, then core melt follows and Sequence S06 ends in core damage state 2D (containment failure prior to core damage. The plant damage state binning is based on the criteria used in NSAC-159).

S07-S10: These end states are similar to S02-S06 except HPCS has failed and RCIC is providing inventory for core decay heat removal. If any one of the containment heat removal systems are successful, Sequences S07, S08, S09, then the end state is OK. If all three fail, then containment failure occurs, RCIC has failed and core damage resulted prior to containment failure. Therefore, Sequence S10 ends in a core damage state 1B (high pressure melt with subsequent containment failure).

S11-S21: If the high pressure systems HPCS and RCIC fail, then the depressurization function is required. Sequences S11, S12, and S13 end states are OK as they represent successful reactor depressurization, inventory control by LPCS or LPCI for core decay heat removal, and success of one of the three containment heat removal functions. If all three containment heat removal functions fail, containment subsequently overpressurizes which makes the SRVs unavailable and the reactor repressurizes. Therefore, the low pressure systems cannot inject, core damage results and the containment fails. Sequence S14 represents this end state as a high pressure melt with subsequent containment failure. Sequences S15, S16, S17 and S18 are the same except core decay heat removal is by the condensate or condensate booster pumps instead of the low pressure injection systems and one of the containment heat removal systems is successful. If the condensate system, as well as the low pressure injection systems, fail with the reactor depressurized, the emergency procedures direct the operator to align standby service water for injection through RHR-B loop. If that is successful, the core decay heat is removed and transferred to the suppression pool via the SRVs. Since the LPCI function is failed, it is assumed containment heat removal via RHR is failed and similarly, since condensate/condensate boosters are failed, containment heat removal via re-establishing the condenser is not available. Therefore, only containment venting is viable for containment heat removal and if successful, the end state is OK (Sequence 19). If containment venting fails, then again the reactor repressurizes which causes loss of the low pressure injection source (standby service water for this sequence) and then containment fails (Sequence 20). If the standby service water cross-tie to the RHR-B loop fails, then core decay heat removal function fails with core damage resulting and containment failure follows (Sequence 21).

S22: If the depressurization function fails with the high pressure injection systems failed, then a high pressure melt ensues with containment failure later.

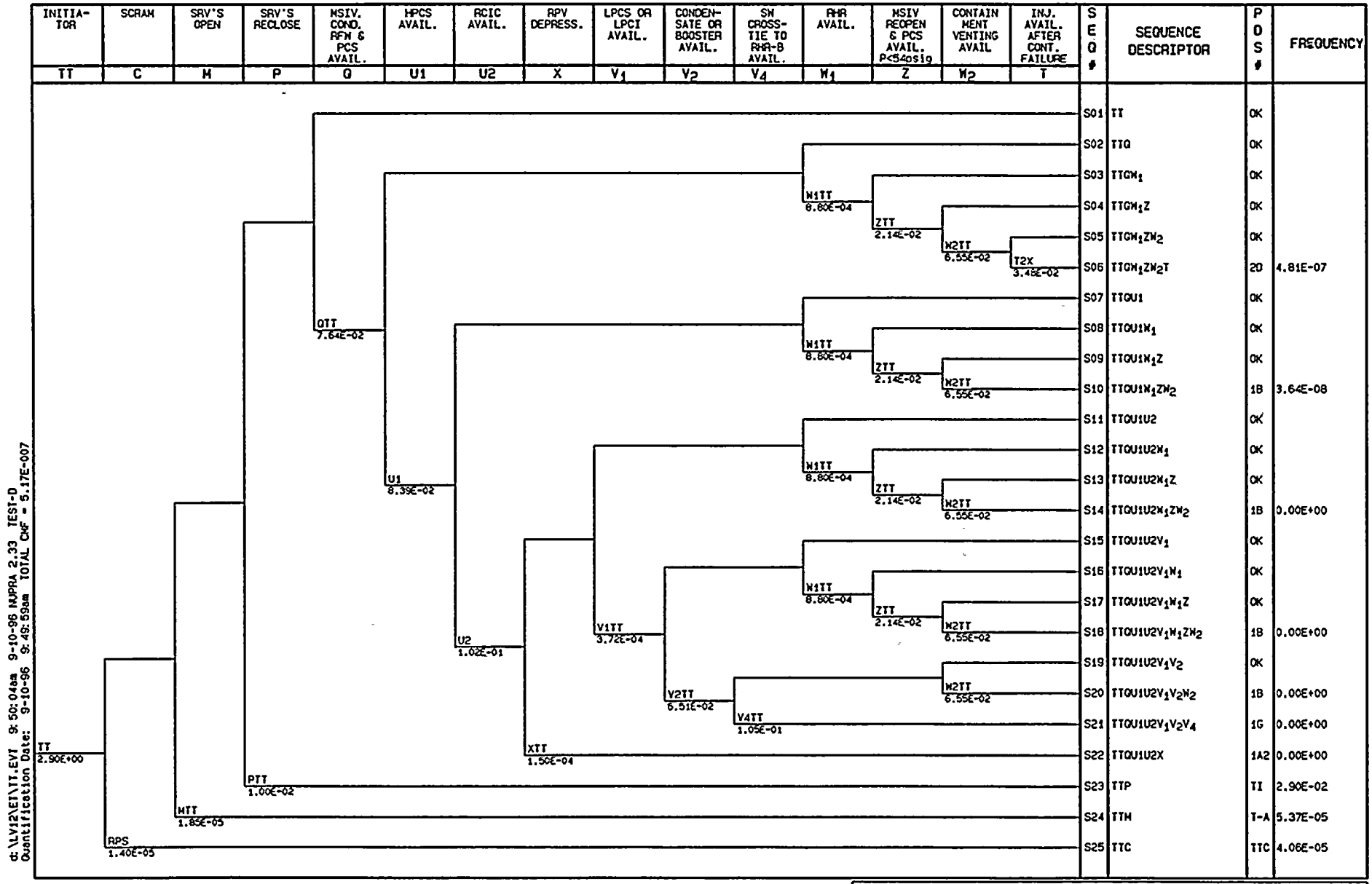
S23-S25: These sequences represent transfers to other event trees for Stuck Open Relief Valve, LOCA, and ATWS respectively.



LOSS OF OFFSITE POWER EVENT TREE
FIGURE 3.1.2.1-7.2

AUTHOR_____DATE_____

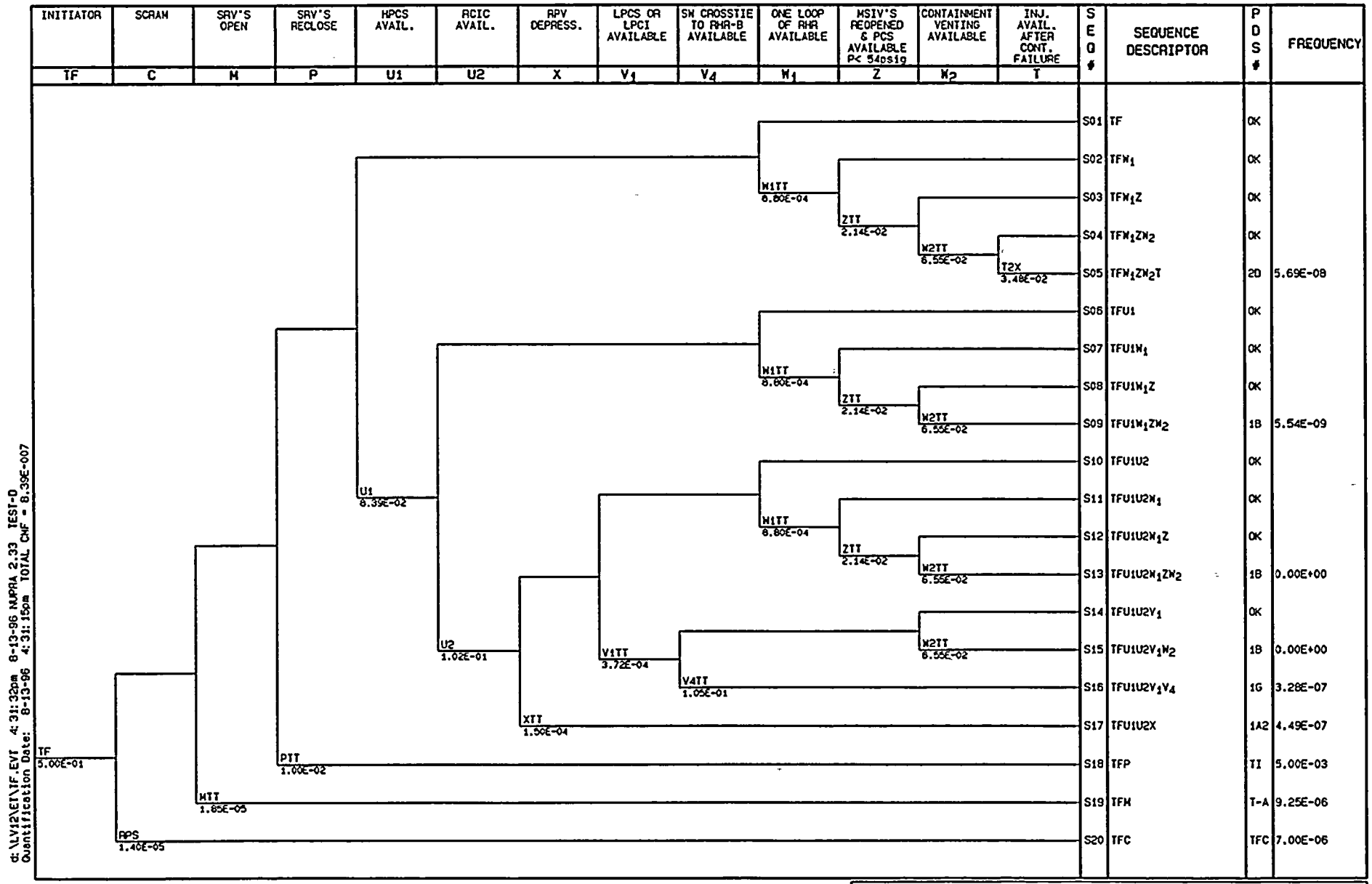
REVIEWER_____DATE_____



TURBINE TRIP EVENT TREE
FIGURE 3.1.2.1-1

AUTHOR_____DATE_____

REVIEWER_____DATE_____



LOSS OF FEEDWATER EVENT TREE
FIGURE 3.1.2.1-4

AUTHOR_____DATE_____

REVIEWER_____DATE_____

3. Alternate Injection Sources

The use of the fire water system for injection to the core was credited on a very limited basis in the licensee's IPE. Specifically, fire water was not credited for station blackout although it has diesel driven pumps independent of AC power. The intent of Generic Letter 88-20, however, was for the licensees to identify instances of plant cost-effective improvements reducing the CDF. Since Station Blackout contributes 61.0 percent to the total CDF, it appears that it would be reasonable for the licensee to consider improvement(s) enabling the use of fire water during station blackout. Please explain why you did not address the use of fire water during station blackout as part of the IPE improvement effort.

RESPONSE:

Revision 2 includes the use of fire protection water in the short term station blackout scenario (see the TE event tree attached to the response for question 2, Branch V_3). As directed by the Emergency Operating Procedures (EOPs), if the reactor water level falls to the low, low (L1) setpoint, the operator will ensure two or more systems are injecting from the preferred list of systems (i.e., Condensate, HPCS, LPCS, RHR-A, RHR-B, RHR-C). If these systems fail, then he will start alternate injection systems such as, RHR/SW Crosstie and/or Fire Protection Water. The Human Reliability Analysis shows that the time required for connecting fire protection water to the condensate system is approximately 70 minutes. For the General Transients and LOCA cases, the diagnosis time for human action starts after water level has dropped below L1 and there has been no successful low pressure injection. Therefore, after the operator realized that the preferred list of systems and RHR/SW Crosstie have failed, there is not sufficient time for him to connect the Fire Protection Water to prevent core damage.

Credit for FP water was only taken in the short-term Station Blackout sequence in which the operator would start to diagnosis the plant conditions at post-scram water level (L3). In this scenario, the operator would have noticed that high pressure injection systems failed to initiate at L2 and knows the other low pressure systems are not available due to blackout conditions. Therefore, there is sufficient time for the operator to diagnosis the problem and make the decision to depressurize and lineup the Fire Protection water. During the long-term SBO scenario, the containment pressure may be too high and preclude the use of the depressurization system (see response to previous question on core cooling/containment heat removal interface). Therefore, the Fire Protection water was not credited for this scenario. As noted in our response to Question #8 of the RAI, the IPE report text is not consistent. The proposed text changes are attached and will clarify the use of Fire Protection water in the PSA.

With clarification of the IPE text, the use of the Fire Protection system as an injection source is consistently handled within the IPE. The current results show that the dominant cause of failure for the Fire Protection system is attributable to human error, i.e., the operating staff fails to connect the Fire Protection Water System to the condensate system. The risk reduction worth of the Fire Protection system is approximately 1.05 and its risk achievement worth is

approximately 1.17. This system becomes important in the short-term SBO only as a backup to the RCIC system. In summary, the dominate cause of failure is attributable to human error, its use is already procedurialized in the emergency operating procedures, and practiced annually. The Supply System will consider further improvement of Fire Protection as an injection source with other cost beneficial insight evaluations from the PSA and in the development of its severe accident management efforts.

TABLE 3.1.1-4

SUCCESS CRITERIA FOR MITIGATING SYSTEMS

INITIATING EVENT	COOLANT INJECTION	CONTAINMENT HEAT REMOVAL
LARGE LOCA	HPCS (or) 1 LPCS Loop (or) 1 LPCI Pump*	1 RHR
INTERMEDIATE LOCA	1 LPCS Loop + 2 SRVs*** (or) 1 LPCI Pump + 2 SRVs*** (or) 1 Condensate Pump + 2 SRVs (or) HPCS (or) 1 FW Pump** FP Water or SW-Crosstie****	1 RHR
SMALL LOCA	1 LPCS Loop + 3 SRVs (or) 1 LPCI Pump + 3 SRVs (or) 1 Condensate Pump + 3 SRVs (or) HPCS (or) RCIC (or) 1 FW Pump (or) FP Water or SW-Crosstie****	1 RHR
TRANSIENT/IORV/SORV	Same as Small LOCA	1 RHR or PCS
ATWS	Dependent on Power Level	PCS

* In the long term (> 2 hours) a different combination may be required.
** MSIVs must be manually reopened, if previously closed, for this system to operate.
*** For liquid breaks less than 0.2 ft^2, 3 SRVs are needed.
**** Capacity sufficient for removal of core decay heat (requires manual setup & initiation).

V_2 - Condensate System Available

Three condensate pumps take suction from the condenser hotwell via a single header. The condensate is directed to the suction of 3 condensate booster pumps. Water can be injected through the feedwater pumps or the bypass line into the reactor at low pressure. One condensate pump or one booster pump is required to maintain reactor water inventory. For maintaining reactor water inventory over extended periods, condenser hotwell makeup from CST is required for the condensate system. The operating range for the condensate pump is from 160 to 0 psig. The operating range for the booster pump is from 560 to 0 psig with a condensate pump operating. A detailed fault tree for the condensate system is developed in a system notebook.

V_3 - FP Water Available

An isolation valve and a fire hose connection are installed on the suction of the "A" condensate booster pump. FP water from two nearby fire hydrants can be injected into the reactor at low pressure. There are 4 fire pumps. Three of these (one diesel and two electric motor driven) take suction from the circulating water basin. The fourth pump (diesel) takes suction from the bladder tank which is filled from either the Well House Storage Tank or from the TMU system. One fire pump is sufficient to maintain reactor water inventory. A detailed fault tree for the FP water injection is developed in its system notebook and the process is detailed in the emergency procedures. The human reliability analysis performed for implementing the Fire Protection water injection feature shows that, for transients and LOCAs where water level may drop to low, low level (L1) before discovery that the other low pressure injection systems are not available, there is not sufficient time to connect and align the Fire Protection Water System. For cases, such as Station Blackout, where the operator knows low pressure systems are not available, then the Fire Protection water is a viable injection source at low reactor pressure.

V_4 - SW Crosstie to RHR-B Available

Service water from the SW B header can be lined up to the discharge of the RHR Heat Exchanger B via 2 keylocked valves. The keys are maintained in the control room under Administrative Control. When the valves are open, service water can be directed to any path associated with the RHR loop B for coolant injection into the core. The human reliability analysis for operator actions required to crosstie SW-B to RHR-B is performed in the human reliability analysis. The SW-B system pressure at the point of injection is 70 psig.

3.2.1.19 FP Water System Description

Fire Protection System Functions

The WNP-2 Fire Protection systems consist of passive and active systems that will detect, extinguish, or contain fire in any fire area. Buildings are divided into fire areas or zones and separated by fire barriers or spatial separation. Fire detection is provided in most areas. These alarms annunciate in the main control room. Fire protection water (Figure 3.2.1-19) is provided by 4 fire pumps and a circular yard loop with sectional control valves. Automatic suppression is provided in areas required as noted in FSAR Appendix F, Fire Hazard Analysis.

The passive systems, i.e., fire walls, fire doors, fire dampers, penetration fire seals and electrical cable barrier protection are used in the safe shutdown analysis. Active systems, wet, pre-action and deluge sprinkler systems, gas (Halon 1301 or CO_2) systems, or manual systems, i.e., hoses stations, hydrants, and extinguishers are not included as part of the safe shutdown system. The fire detection and alarm system is used for the early notification of a fire, but they also are not used for the safe shutdown analysis.

In summary, the active or alarm systems are not used for safe shutdown and thus total loss will not affect the ability of the plant to safely shut down due to a fire. However, the passive parts of the fire protection system are needed to limit the spread of a fire and assure the plant can be safely shut down. The system function during a fire will be addressed in the WNP-2 IPEEE.

Emergency Water Makeup Function

There are two water supplies for the fire protection system; 3 fire pumps taking suction from the Circulating Water Pump House (CWPH) intake basin and one fire pump taking suction from a 400,000 gallon bladder tank. The CWPH intake basin is considered to be an unlimited water source as the maximum refill is 25,000 gpm while maximum calculated evaporation rate of the cooling towers is about 18,500 gpm. The second source (the 400,000 gallon bladder tank) can be refilled in 8 hours or less.

The Fire Protection system can supply water to the RPV under emergency conditions. The WNP-2 emergency procedures provide a step-by-step procedure to connect the fire water system to the suction on condensate pump 2A. ~~However, because of timing constraints, credit for this function was not taken in the IPE.~~ This procedure was verified and validated by plant walk throughs when it was first established as part of the Emergency Operating Procedures and has been practiced by the Equipment Operators on a periodic schedule. All of the equipment needed using the Fire Protection system in emergency core cooling mode (i.e., hoses, spanner, shutoff valves, etc.) are dedicated for this purpose and are permanently placed in the plant at or near the location where they would be used. Such emergency procedure equipment is clearly marked for the sole use of implementing the emergency procedures.

on the instrumentation (Matrix 1). The LPCI-A, B and C are partially dependent on the ADS I and II, the Suppression Pool Cooling, DW Coolers, DW Sprays, WW Sprays, and the Venting (Matrix 4).

- CONDENSATE SYSTEM

The condensate is a low pressure system. Three condensate pumps take suction from the condenser hotwell via a single header. They are motor driven pumps with the power supply from the offsite power. The motors are cooled by a combination of air and water. Water from the TSW is utilized to cool the upper thrust bearing of the motors. After passing through gland seal steam condenser, SJAE condensers, offgas condenser and the filter demineralizers, the condensate is directed to the suction of 3 condensate booster pumps. They are motor driven pumps with the power supply from the offsite power. The pumps are oil and water cooled. The water is from the TSW. Two lubricating oil systems are also cooled by the TSW. After passing through 5 series of heaters, the condensate is directed to the suction of the RFW system. TSW depends on Division 1 or 2 AC and Division 1 or 2 DC. RFW depends on CAS for startup valve control. Therefore, the condensate system is dependent on the offsite power, and the TSW, and is partially dependent on the Division 1 and 2 AC, and Division 1 and 2 DC (Matrix 1). The condensate system is partially dependent on the ADS I and II, the condenser and the RFW (Matrix 4).

- FP WATER

An isolation valve and fire hose connection are installed on the suction of the 'A' condensate booster pump. There are 4 fire pumps. Three of these (one diesel and two motor driven) take suction from the circulating water basin. The fourth pump (diesel) takes suction from the bladder tank which is filled from either the well house storage tank or from the TMU system. ~~Since the FP water has to go through the COND and the RFW, it has the same dependencies as the COND and the RFW except for the offsite power and the TSW. Therefore,~~ the FP water is partially dependent on the offsite power for its non-diesel powered motor driven fire pumps (Matrix 1) and to operate valves in the Condensate/Feedwater flow path in a timely manner (the injection alignment can be accomplished manually without AC power dependency). Since the FP water is a low pressure system, it partially depends on ~~RFW, and~~ ADS I and II for depressurization before injection can occur (Matrix 4).

- DW COOLERS

Cooling of the drywell is provided by 5 fan coil units which recirculate containment air through cooling coils. Heat is transferred to the RCC system. The RCC system dumps its heat to the TSW system. RCC isolates on a LOCA signal. CRA-FNS-1A2, 1B2, 1C2, 2A1, 2B1, 4A and 4B all get a start signal to serve as drywell air mixers on F and A signals. There are 2 fans that draw air from the containment head area and return it to the general drywell area. In addition, there are 7 recirculation fans that provide recirculation of air in the drywell. Some fan coil units and fans depend on Division 1 AC; and others on Division 2 AC. During normal operation, 5