Progress Energy

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261 / LICENSE NO. DPR-23

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING SEVERE ACCIDENT MITIGATION ALTERNATIVES ANALYSIS

Ladies and Gentlemen:

By letter dated June 14, 2002, Carolina Power & Light (CP&L) Company submitted an application for the renewal of the Operating License for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, also referred to as RNP.

By letter dated October 23, 2002, the NRC provided a request for additional information to CP&L regarding the Severe Accident Mitigation Alternatives analysis contained in the Environmental Report. The response to the request for additional information is contained in the Attachments to this letter. Note, however, that the response to NRC Request 9 will be delayed as discussed in a telephone call between CP&L and NRC on December 23, 2002. CP&L will provide a schedule for providing Response 9 by January 15, 2003.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

B. L. Fletcher III
Manager - Support Services - Nuclear

Attachments:

I. Affirmation
II. Response to Request for Additional Information Regarding Severe Accident Mitigation Alternatives
III. Appendix A from Probabilistic Safety Assessment Summary Document -1997
IV. Calculation RNP-F/PSA-0001, without Attachments

A001

JSK/jsk

c: Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC) (w/o Attachments)
Mr. L. A. Reyes, NRC, Region II (w/Attachments)
Mr. R. Subbaratnam, NRC, NRR (w/o Attachments)
NRC Resident Inspectors, HBRSEP (w/o Attachments)
Attorney General (SC) (w/o Attachments)
Mr. S. K. Mitra, NRC, NRR (w/Attachments)
Mr. R. L. Emch, NRC, NRR (w/Attachments)
Mr. R. M. Gandy, Division of Radioactive Waste Management (SC) (w/o Attachments)

AFFIRMATION

The information contained in letter RNP-RA/02-0180 is true and correct to the best of my information, knowledge, and belief, and the sources of my information are officers, employees, contractors, and agents of Carolina Power and Light Company. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 2 JAN 03

Timothy P. Cleary
T. P. Cleary
Plant General Manager, HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING SEVERE ACCIDENT MITIGATION ALTERNATIVES

NRC Request 1:

"The SAMA analysis is based on the most recent version of the RNP Probabilistic Safety Assessment (PSA) model for internal events (i.e., the MOR99 model), which is a modification to the original Individual Plant Examination (IPE) developed in 1992 and the updated PSA developed in 1997. Please provide the following information regarding this PSA model:

a. a summary description of the internal and external peer reviews of the level 1, 2, and/or 3 portions of this PSA,

b. a characterization of the findings of the Westinghouse Owners Group peer review conducted in 2001, and the impact of any identified weaknesses on the SAMA identification and evaluation process,

c. a description of the major differences from the IPE model, including the plant and/or modeling changes that have resulted in the new core damage frequency (CDF) and the large early release frequency (LERF),

d. a breakdown of the internal event CDF and LERF by major contributors, in a format similar to that used in either the IPE or the 1997 PSA summary report,

e. a breakdown of the population dose (person-rem per year within 50 miles) by containment release mode in the following form, or equivalent:

Containment Release Mode	Fraction of Population Dose
SGTR	
Interfacing Systems LOCAs	
Containment isolation failure	
Early containment failure	
Late containment failure	
No containment failure	

f. for each containment release category (including LERF and non-LERF contributors): the associated release frequency, release magnitude (fractions), and MACCS-calculated conditional consequence measures (where available). Please identify those release categories that are considered to contribute to LERF, and those categories to which SGTR and ISLOCA releases are assigned,

g. justification for neglecting large late release categories in establishing the baseline estimate of offsite consequences, given that large late releases could result in population doses comparable to those for large early releases. Include a justification for not using RC-1A and/or RC-1BA to represent large late releases, given that these release categories result in greater releases of volatile fission products and potentially greater releases of non-volatile fission products than RC-1B,

h. the definition of LERF used to distinguish a large-early release from a small-early or a large-late release, and

i. clarification of whether the reported CDF and LERF is per reactor year or per calendar year."

CP&L Response 1.a:

"a summary description of the internal and external peer reviews of the level 1, 2, and/or 3 portions of this PSA,"

The H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, also referred to as RNP, IPE and Probabilistic Safety Assessment (PSA) have been subjected to a number of assessments and reviews. The following peer reviews have been performed:

1989: *External Peer Review of The H.B. Robinson Unit 2 Level 1 Probabilistic Risk Assessment, Pickard, Lowe and Garrick, Inc.* This review was performed by J. W. Stetkar, Pickard, Lowe, and Garrick, Inc.; Michael V. Frank, Safety Factor Associates; W. J. Parkinson, Science Applications International Corporation (SAIC); G. W. Parry, NUS; and R. L. Summitt, SAROS. The review was performed in general accordance with NSAC/67 and it included evaluation of the overall structure of the plant model, the bases, assumptions, and models for the dominant core damage contributors, and the methodology for evaluating post-initiator operator actions. Reviewers provided insights to help Carolina Power & Light (CP&L) Company adapt the PSA for submittal to NRC under the IPE program.

1991-1992: As indicated in Section 5.0 of the IPE, inputs to and outputs from the IPE analysis were reviewed and evaluated by CP&L's Nuclear Fuels Section, who performed RELAP analyses of a plant specific RNP model to validate success criteria, and personnel from operations, training, the plant simulator, licensing, engineering and other organizations.

1996: *Updated Final Report for the Independent Peer Review of the H. B. Robinson PSA Model,* R. Anoba, SAIC. This review compared the IPE model with the then current PSA model, and evaluated model and logic changes between the two. It evaluated the overall PSA methodology in general and focused closely on the quantification methodology, and it identified potential model updates for consideration. This review focused on the Level 1 model.

2001: *Westinghouse Owners Group (WOG) Risk Based Technology Working Group (RBTWG) Peer Certification Review.* A comprehensive review of the Level 1 and Level 2 models was performed by L. Kachnik, South Carolina Electric & Gas (SCE&G); R. Lichtenstein, TXU; R. Bertucio, Scientech; S. Rodgers, Erin; and B. Sloane and R. Lutz, Westinghouse.

ERIN Engineering performed most of the Severe Accident Mitigation Alternatives (SAMA) analysis in support of the Environmental Report (ER). ERIN's effort involved the following key tasks:

1. Development of a list of SAMA candidates based on past experience and plant specific insights.

2. Calculation of the maximum averted cost-risk.

3. Quantification of the PSA model to represent proposed plant modifications.

4. Calculation of cost benefit related to the plant modifications.

ERIN has performed internal independent reviews of each analysis task. This internal review involved an assessment of the methods used in the analysis, a review of the key assumptions, and a check of the calculated results by a qualified independent reviewer. In addition, the reviewer compared the calculated results with those available from similar projects to assure consistency.

Tetra Tech NUS (TtNUS) performed the SAMA Level 3 modeling using the Melcor Accident Consequence Code System (MACCS) 2. TtNUS performed no peer review of the developed model, but subjected its use of the model to the TtNUS quality assurance procedure for performing technical work. A qualified independent technical analyst reviewed the work and TtNUS line management approved performance of the review.

CP&L provided technical direction to ERIN and TtNUS and reviewed the analytical results, providing comments and direction as appropriate.

CP&L Response 1.b:

"a characterization of the findings of the Westinghouse Owners Group peer review conducted in 2001, and the impact of any identified weaknesses on the SAMA identification and evaluation process,"

The WOG peer review of the Robinson PSA was conducted in November 2001. A draft report summarizing the results of the review has been received, but has not yet been finalized. The RNP peer review results demonstrate that the RNP PSA model is of appropriate quality for SAMA analyses. The results of the peer review are characterized in the following table:

PSA Element	Assigned Grade
Initiating Events	3
Accident Sequence Evaluation	3(C)
Thermal Hydraulic Analysis	3(C)
System Analysis	3
Data Analysis	3(C)
Human Reliability Analysis	3
Dependencies	3
Structural Response	3
Quantification	3(C)
Containment Performance	3
Maintenance and Update	3

A grade of "3" is defined in the draft report as follows:

"This grade extends the requirements to assure that risk significance determinations made by the PRA are adequate to support regulatory applications, when combined with deterministic insights. Therefore, a PRA with elements determined to be at Grade 3 can support physical plant changes when it is used in conjunction with other deterministic approaches that ensure that defense-in-depth is preserved. Grade 3 is acceptable for Grade 1 and 2 applications, and also for assessing safety significance of equipment and operator actions. This assessment can be used in licensing submittals to NRC to support positions regarding absolute levels of safety significance if supported by deterministic evaluations."

The draft report also indicates the following in reference to the contingent grades, noted as "3(C)": "Grades assigned contingent upon addressing certain comments or recommendations from the review are noted using a 'C'."

All but one of the comments received for contingent grades in the draft report were at or below the "B" significance level. A "B" significance level is defined in the draft report as "Important and necessary to address, but may be deferred until the next PRA update. . ." Therefore, only the "A" significance level findings need to be evaluated and potentially addressed before the next regular PSA update. One "A" significance level finding was provided for the Robinson PSA, regarding the quantification element. The discussion of this finding indicates that "the core damage frequency model is presently quantified at a cutoff of 4.00E-09. Many PRAs are quantified using a much lower cutoff..."

The cutoff value employed in the Robinson PSA is more than four orders of magnitude below the calculated baseline core damage frequency and is consistent with the guidance provided in Electric Power Research Institute (EPRI) Technical Report TR-105396.

A further reduction in truncation level does not impact the SAMA identification and evaluation process. The SAMA process concerns itself with identifying candidate plant or procedure changes that have the highest potential for reducing core damage frequency and person-rem, and with determining whether or not the implementation of those candidates is beneficial on a cost-risk reduction basis. By definition, truncated cutsets are very low-probability contributors.

Therefore, the current Robinson PSA is appropriate for use in identification and evaluation of potential SAMAs.

The "A" and "B" significance level findings have been entered into the CP&L corrective action program for evaluation and disposition.

CP&L Response 1.c:

"a description of the major differences from the IPE model, including the plant and/or modeling changes that have resulted in the new core damage frequency (CDF) and the large early release frequency (LERF),"

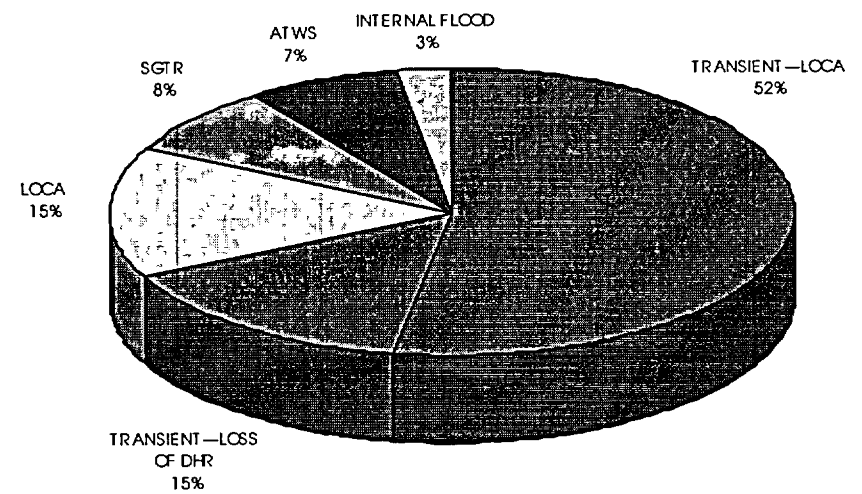
The changes from the IPE model are described in Appendix A from the 1997 PSA Summary Document, and in calculation RNP–F/PSA–0001. The documents are provided as Attachments III and IV to this response.

CP&L Response 1.d:

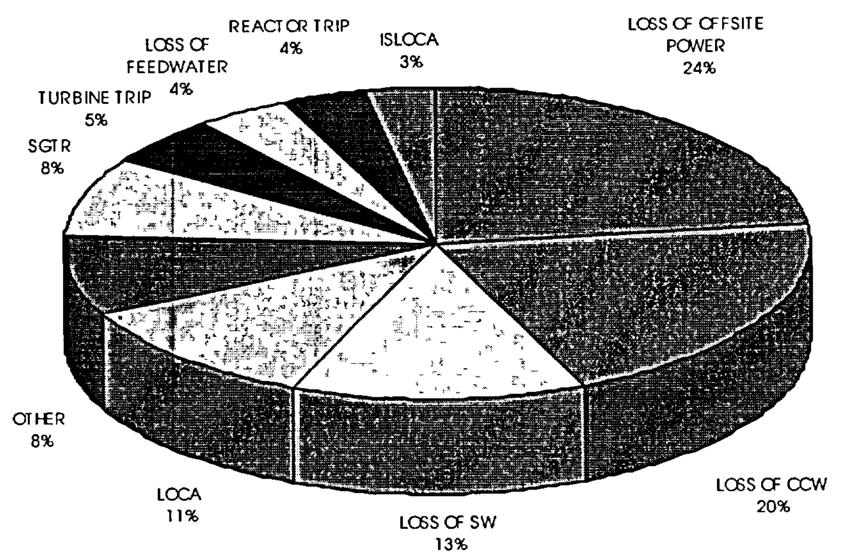
"a breakdown of the internal event CDF and LERF by major contributors, in a format similar to that used in either the IPE or the 1997 PSA summary report,"

The follow figures are updated versions of the figures in the 1997 PSA Summary Document.

Contribution to CDF by Sequence Type



Contribution to CDF by Initiator



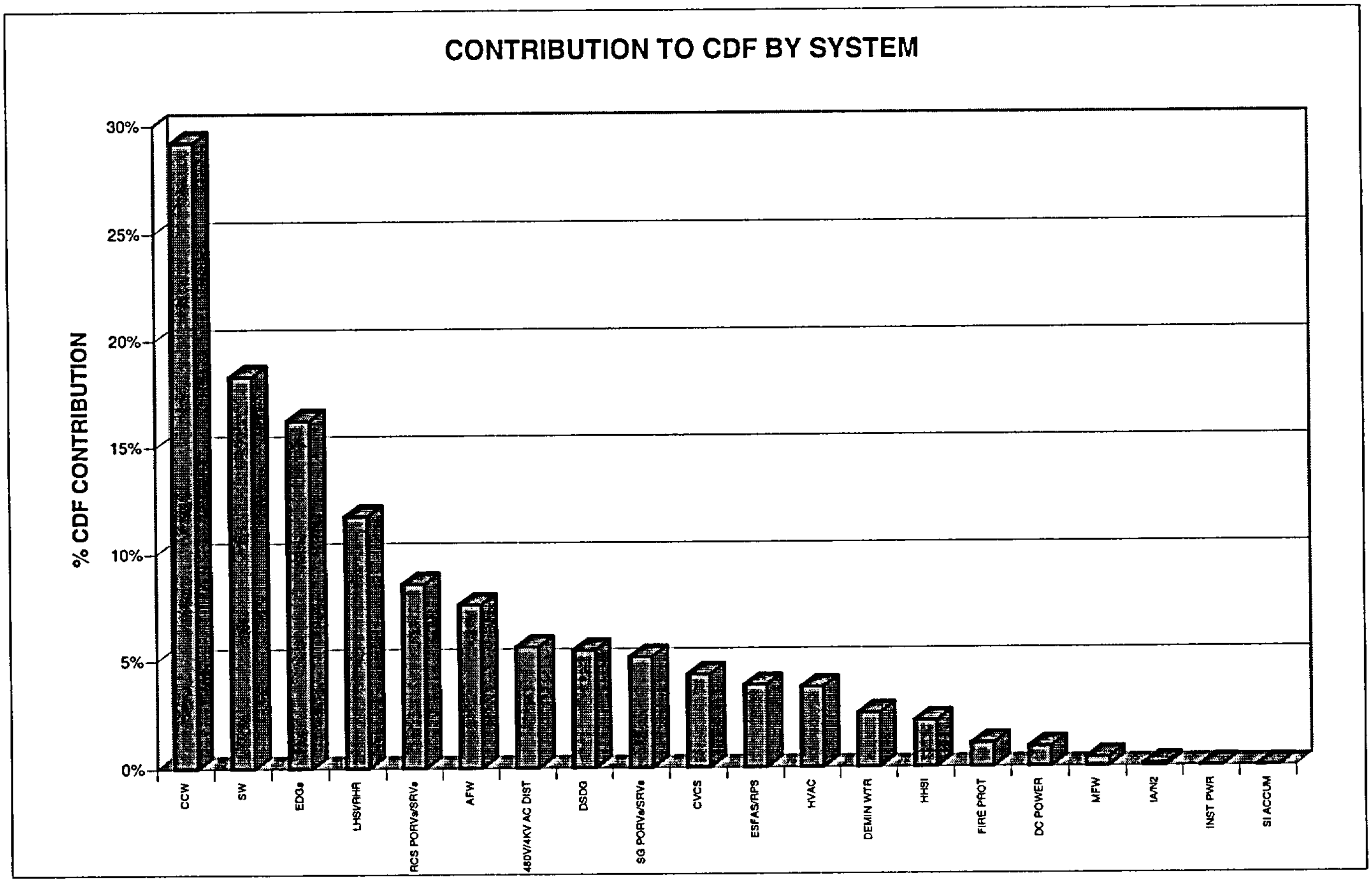

CONTRIBUTION TO CDF BY SYSTEM
% CDF CONTRIBUTION
30%
25%
20%
15%
10%
5%
0%
CCW
SW
EDGs
LHSI/RHR
RCS PORVs/SRVs
AFW
480V/4KV AC DIST
DSDG
SG PORVs/SRVs
CVCS
ESFAS/RPS
HVAC
DEMIN WTR
HHSI
FIRE PROT
DC POWER
MFW
IA/N2
INST PWR
SI ACCUM

Table 1.d-1

Top 50 Component Importances - Normalized

Basic Event	DESCRIPTION	Relative Importance
KCCF%RUN	COMMON CAUSE FAILURE (CCF) OF ALL CCW PUMPS TO RUN	100 0
#ACBCRDCC	CCF OF REACTOR TRIP BREAKERS	60 4
KRV%729NN	RELIEF VALVE CC-729 TRANSFERS OPEN AND DIVERTS FLOW	59 0
PCCFFOTPLN	CCF OF FUEL OIL TRANSFER PUMPS AND VALVES	52 1
WCCF%ABCD	CCF TO RUN ALL SW (SERVICE WATER) PUMPS	46 0
RPVCV456FF	PORV PCV-456 FAILS TO RECLOSE AFTER DEMAND	32 4
RPVV455CFF	PORV PCV-455C FAILS TO RECLOSE AFTER DEMAND	32 4
KPM%CCWBKR	CCW PUMP B FAILS TO RUN FOR A YEAR	31 8
UTMDGDSSDG	DEDICATED SHUTDOWN DIESEL GENERATOR UNAVAILABLE	30 9
FPT1XSABFR	TURBINE-DRIVEN PUMP FAILS TO RUN	29 1
PTMDGEDG-B	EMERGENCY DIESEL GENERATOR B UNAVAILABLE	28 7
KCCFRUN	CCF OF ALL CCW PUMPS TO RUN	26 3
PDFFOTPBNN	MOTOR-DRIVEN FUEL OIL TRANSFER PUMP B FAILS TO START	23.8
PTMDGEDG-A	EMERGENCY DIESEL GENERATOR A UNAVAILABLE	20 7
JTMCHGPMPA	CHARGING PUMP TRAIN A UNAVAILABLE	20 0
KMVC749BTN	MOV(STANDBY) CC-749B FAILS TO OPEN	19 1
KMVC749ATN	MOV(STANDBY) CC-749A FAILS TO OPEN	19 1
PDFFOTPANN	MOTOR-DRIVEN FUEL OIL TRANSFER PUMP A FAILS TO START	19 1
FTMSDPTRXM	AFW STEAM DRIVEN PUMP TRAIN C UNAVAILABLE	19 0
FPT1XSABFS	TURBINE-DRIVEN PUMP FAILS TO START	18 8
LMVS862AOP	MOV OPERATOR SI-862A FAILS (STANDBY)	16 4
LMVS862BOP	MOV OPERATOR SI-862B FAILS (STANDBY)	16 1
JPM%CHPCJR	CHARGING PUMP C FAILS TO RUN	15 8
JPM%CHPBJR	CHARGING PUMP B FAILS TO RUN	15 8
NTMDSBUS	DS BUS UNAVAILABLE	15 0
ZCCFDGIFTS	CCF OF 2 OF 2 INLET FANS IN THE EDG ROOMS TO START	12 8
ZCCFDGEFTS	CCF OF 2 OF 2 EXHAUST FANS IN THE EDG ROOMS TO START	12 8
QPVRV1-3NN	PORV RV1-3 FAILS TO OPEN	12 6
QPVRV1-2NN	PORV RV1-2 FAILS TO OPEN	12 6
QPVRV1-1NN	PORV RV1-1 FAILS TO OPEN	12 6
KCCF%ACFTR	COMMON CAUSE FAILURE OF CCW PUMPS A&C TO RUN	12 0
ETD2/27BNN	TIME DELAY RELAY 2/27B FAILS TO ENERGIZE	11 3
ECCFDGTIME	CCF OF THE TIME DELAY RELAYS FOR DIESEL GENERATOR ACTUATION	10 5
JFLSEALIFN	SEAL INJECTION FILTER CLOGS	10 3
UDFDSFOPNN	MOTOR-DRIVEN DS FUEL OIL PUMP FAILS TO START	10 2
WMVV616CFF	MOV V6-16C FAILS TO CLOSE ON DEMAND	10 0
PCCFEDG/CB	EDG COMMON CAUSE FAILURE TO START MODULE	10 0
UDGDSSDGDR	DIESEL GENERATOR DS FAILS TO RUN	10 0
PDGEDG-BNN	DIESEL GENERATOR EDG-B FAILS TO START	9 7
KCCFABCFTS	CCF OF ALL CCW PUMPS TO START OR CV CC-702A&B&C TO OPEN	9 3
LCCF862FTC	SI-862A/862B COMMON CAUSE FAIL TO CLOSE OF MOTOR OPERATOR	9 1
ETD2/17BNN	TIME DELAY RELAY 2/17B FAILS TO ENERGIZE	8 9

Table 1.d-1

Top 50 Component Importances - Normalized

Basic Event	DESCRIPTION	Relative Importance
#ACRDMF	CONTROL RODS FAIL DUE TO MECHANICAL BINDING	8.9
VEPFWED/FN	DIESEL-DRIVEN FIRE PUMP FAILS TO RUN	8.6
NTMCB52/7	CIRCUIT BREAKER (CB) 52/7 UNAVAILABLE DUE TO TESTING/MAINTENANCE (T/M)	8.1
NTMCB52/12	CB 52/12 UNAVAILABLE DUE TO T/M	8.1
WTMNORTHDR	SW NORTH HEADER UNAVAILABLE (STRAINER)	8.1
KXVC794AFN	MANUAL VALVE CC-794A TRANSFERS CLOSED	8.1
KXVC728DFN	MANUAL VALVE CC-728D TRANSFERS CLOSED	8.1
PDGEDG-ANN	DIESEL GENERATOR EDG-A FAILS TO START	7.9

Table 1.d-2

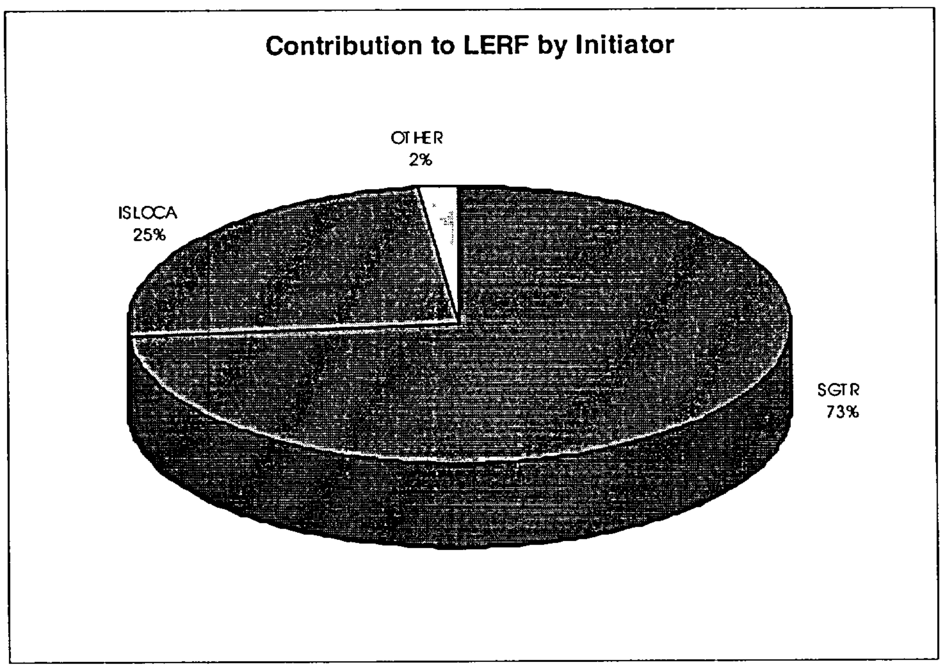
Operator Action Importance – Normalized

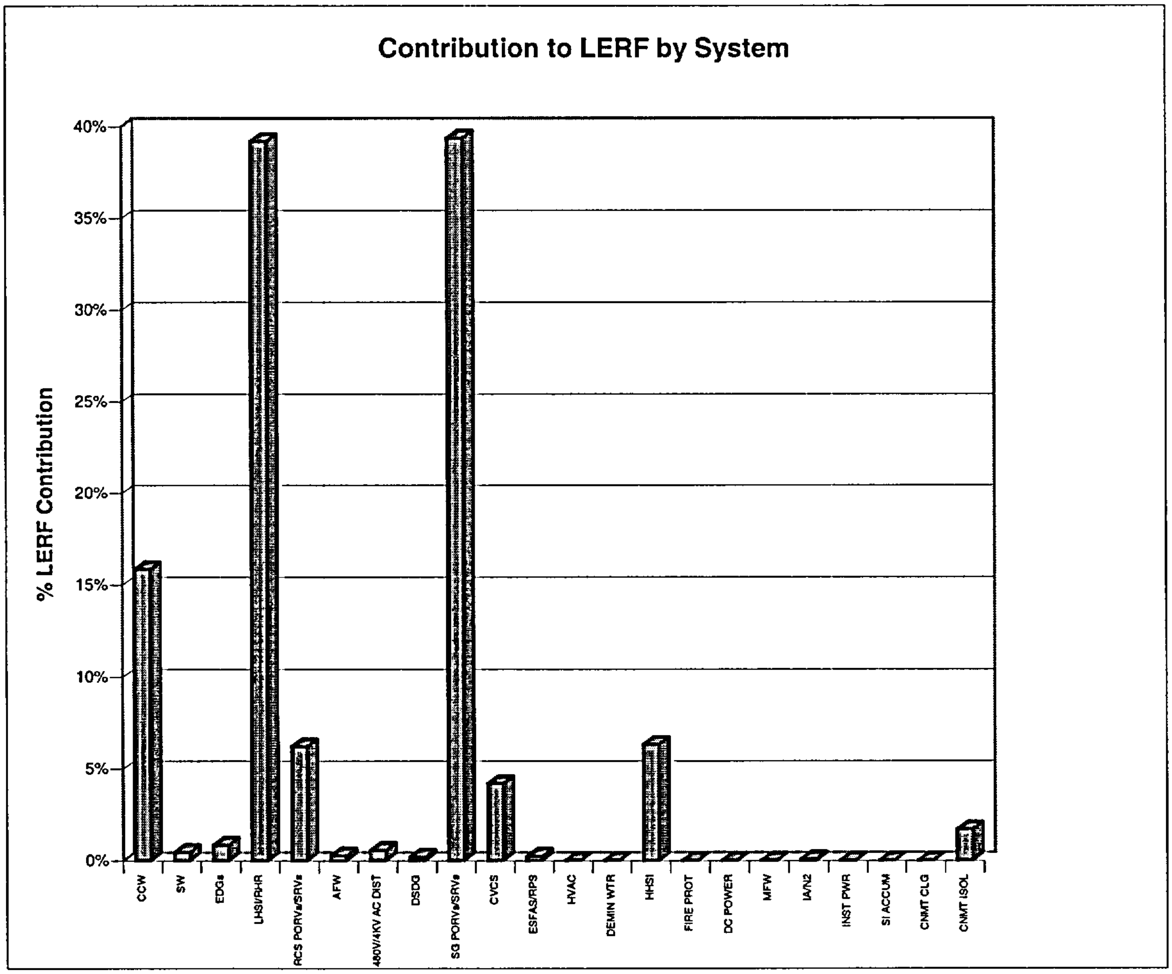
Basic Event	DESCRIPTION	Relative Importance
OPER-4	OPERATOR FAILS PROVIDE ALTERNATE COOLING TO CHARGING PUMPS	100.0
OPER-11	OPERATOR FAILS TO IDENTIFY/ISOLATE SW PIPE RUPTURE	23.2
OPER-18B	OPERATOR FAILS TO SUPPLY AFW WITH SW	19.4
OPER-18A	OPERATOR FAILS TO SUPPLY AFW WITH DEEPWELL PUMPS	17.2
OPER-1	OPERATOR FLAG - FAILURE TO SWITCHOVER TO COLD LEG RECIRCULATION	13.7
OPER-BC	OPERATOR FAILS TO CLOSE INPUT BREAKER TO BATTERY CHARGER FOLLOWING UNDERVOLTAGE ON E1/E2	10.5
OPER-MFBYP	OPERATOR FAILS TO MANUALLY OPEN BYPASS VALVES (FRP-H 1)	9.0
OPER-12	OPERATOR FAILS TO CONTROL AFW STEAM DRIVEN PUMP	8.0
OPER-3	OPERATOR FLAG - FAILURE TO IMPLEMENT BLEED AND FEED	7.8
OPER-10	OPERATOR FAILS TO UTILIZE DEDICATED SHUTDOWN DIESEL GENERATOR	7.5
OPER-S862	OPERATOR FAILS TO LOCALLY CLOSE MOV SI-862A OR B	7.0
OPER-SD	OPERATOR FAILS TO ESTABLISH SHUTDOWN COOLING	4.9
OPER-DE	OPERATOR FAILS TO DEPRESSURIZE USING SG PORVs	3.9
OPER-5	OPERATOR FAILS TO THROTTLE SW TO ONE CCW HX	2.2
OPER-80	OPERATOR FAILS TO PROVIDE LONG TERM RCS MAKEUP	2.1
OPER-6	OPERATOR FAILS TO PROVIDE ALTERNATE COOLING TO AFW PUMPS	2.1
OPER-7	OPERATOR FLAG - FAILURE TO SWITCHOVER TO COLD LEG RECIRCULATION	1.1
OPER-26	OPERATOR FAILS TO ISOLATE TURBINE BLDG. LOADS	0.6
OPER-ALTSW	OPERATOR FAILS TO PROVIDE ALTERNATE COOLING GIVEN SW FAILURE	0.5
OPER-MFW2	OPERATOR FAILS TO ESTABLISH MFW WITHOUT SI INITIATION	0.4
OPER-MCC5	OPERATOR FAILS TO SWITCH SOURCE TO DS BUS	0.3
OPER-J02	CREW FAILS TO ALIGN CHARGING PUMP SUCTION TO RWST	0.2
OPER-25D-1	OPERATOR FAILS TO START SW PUMP D	0.2

Table 1.d-2

Operator Action Importance – Normalized

Basic Event	DESCRIPTION	Relative Importance
OPER-17B	OPERATOR FAILS TO START CCW PUMP B	0.2
OPER-17C	OPERATOR FAILS TO START CCW PUMP C	0.2
OPER-J01	OPERATOR FAILS TO START PUMP AFTER LOSP/PUMP FAILURE	0.1
OPER-SGDN	FAILURE TO RECOVER SG PORVS USING STEAM DUMP N2 ACCUMULATOR	0.1
OPER-43	OPERATOR FAILS TO ESTABLISH EMERGENCY BORATION (FRP-S.1 STEP 4)	0.1





CP&L Response 1.e:

"a breakdown of the population dose (person-rem per year within 50 miles) by containment release mode. . ."

The population dose-risk for the RNP SAMA analysis is determined based on a specific set of release categories that are used in the plant's current PSA model of record (MOR99). The original RNP SAMA submittal included only contributions from those release categories defined as Large Early Release Frequency (LERF) scenarios by the PSA; however, this response to the Request for Additional Information (RAI) provides the population dose-risk for all release categories (both LERF and non-LERF). In addition, the contributions of the following categories to the population dose are provided:

- Steam Generator Tube Rupture (SGTR)
- Intersystem Loss of Coolant Accident (ISLOCA)
- Containment Isolation Failure/Early Containment Failure*
- Late Containment Failure
- No Containment Failure

*Due to the quantification method used in the model, these categories are reported together.

The definitions of the release categories used in the RNP model are provided in Table 1.e-1 for ease of reference. Table 1.e-2 summarizes the dose-risk results. Note that the LERF release categories were used as the sole contributors to the dose-risk in the RNP submittal. These release categories contribute 54.7% of the total dose-risk.

Table 1.e-1: RNP Release Category Definitions

Containment Intact (IC-1)	This release category represents an accident sequence in which the containment is intact. The source term for this type of sequence is very small and limited to the containment design leakage rate.
Release Category 1 (RC-1)	This release category is a late containment failure caused by gradual overpressurization. The core debris is assumed to be coolable. This type of gradual pressure increase is assumed to result in a benign containment failure and the duration of the release could be over a long period of time. The release from containment is scrubbed by either the containment sprays or a pool of water over the core debris.
Release Category 1A (RC-1A)	This release category is similar to RC-1 except that re-vaporization occurs. Re-vaporization is caused by the self-heating of radionuclides plated out on the reactor coolant system becoming re-suspended in the containment atmosphere. This re-vaporization is postulated to occur late in the accident sequence after the containment has failed. This allows the radionuclides to be released from the containment after only a limited holdup time. The impact of re-vaporization on the source term is to increase the contribution of volatile radionuclides to the source term.

Table 1.e-1: RNP Release Category Definitions	
Release Category 1B (RC-1B)	This release category is similar to RC-1 except that no scrubbing by containment sprays and/or water pools is available. If containment sprays function, or the Refueling Water Storage Tank (RWST) inventory is otherwise transferred into containment, then both debris cooling and scrubbing will initially be attained. Un-coolable debris is assumed to eventually exist in these cases due to boil off of any water successfully injected into containment. Thus, this category implies a debris bed which eventually dries up resulting in considerable core-concrete interaction (CCI).
Release Category 1BA (RC-1BA)	This release category is similar to RC-1 except that both re-vaporization and no containment scrubbing are assumed to occur.
Release Category 2 (RC-2)	This release category represents a large, early containment failure. The debris is assumed to be coolable. The large failure significantly reduces the holdup time in the containment. The RNP-specific liner failure releases are assumed to belong to this category. The release from the containment is scrubbed by containment spray operation following fission product releases from the primary side. In this case, the releases will be driven by the prompt release of fission products at containment failure. The effects of re-vaporization, if any, should be small. Thus, release categories with re-vaporization will not be postulated for the large, early containment failures. However, care will be taken when assigning source terms to pick a representative sequence for RC-2 (and RC-2B) that exhibits re-vaporization.
Release Category 2B (RC-2B)	This release category is similar to RC-2 except that no scrubbing by containment sprays and/or water pools is assumed to occur.
Release Category 3 (RC-3)	This release category represents an early containment isolation failure with a small leakage rate (<4" diameter). The core debris is assumed to be coolable. The release from the containment is scrubbed by either the containment sprays or a pool of water over the core debris. For the larger of the small leakage failures (i.e., close to 4" in diameter) the releases will be driven by the prompt release of fission products at containment failure and the effect of re-vaporization.
Release Category 3B (RC-3B)	This release category is similar to RC-3 except that no scrubbing by containment sprays and/or water pools is assumed to occur.

Table 1.e-1: RNP Release Category Definitions	
Release Category 4 (RC-4)	This release category represents a containment bypass accident sequence with a small leakage rate. The leakage rate that would correspond to an SGTR sequence with cycling Safety Relief Valves (SRVs), or an ISLOCA in which operators react in time to mitigate effects by closing the valves on the Residual Heat Removal (RHR) suction line. The core debris is assumed to be coolable and releases from the containment scrubbed. Scrubbing by water in the affected Steam Generator (SG) above the break is assumed to occur. Note that the operating procedures direct the operator to isolate the affected SG. Thus, the affected SG will be dry in the majority of the cases and no fission product scrubbing would occur. This category has been retained for future use, but for the purposes of this study, the unscrubbed source term (RC-4C) is assigned to these low probability branches.
Release Category 4C (RC-4C)	This release category is similar to RC-4 except that no scrubbing by water in the affected SG above the break occurs. The core debris is assumed to be coolable and releases from the containment scrubbed. Note that a release category for no scrubbing by containment sprays and/or water pools is not postulated in this case. This is because, for the bypass sequences, most of the release would be directly from the primary to the environment or the auxiliary building. Re-vaporization is also assumed to be negligible as compared to the direct releases.
Release Category 5 (RC-5)	This sequence represents a containment bypass accident with a large leakage rate. Such a rate is representative of an SGTR accident with a stuck open SRV in the affected SG, or the unmitigated ISLOCA accident. The core debris is assumed to be coolable and releases from the containment scrubbed. The releases from the affected SG are assumed to be scrubbed by water above the break line. However, the probability of scrubbed releases is very small due to present procedures. Thus, similarly to RC-4, the unscrubbed source term (RC-5C) will be conservatively assigned to these low probability branches.
Release Category 5C (RC-5C)	This release category is similar to RC-5 except that no scrubbing by water in the affected SG above the break occurs. The core debris is assumed to be coolable and releases from the containment are scrubbed.

Table 1.e-2: Dose-Risk Results

Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C	
	Non-LERF							LERF						Sum of annual risk
Population dose-risk (person-rem) 0-50 miles	0 10	2 18	0 30	2 01	0 16	0 10	0 00	0 02	0 28	0 00	1 56	3 04	0 94	10 68
Population dose risk (percent) 0-50 miles	0 95	20 40	2 76	18 81	1 50	0 91	0 01	0 22	2 61	0 00	14 60	28 45	8 78	100 00
SGTR %	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	100 00	0 00	82.90	21 88
ISLOCA %	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0.00	0 00	0 00	0 00	100 00	17 10	29.95
Early containment failure and containment isolation failure %	0 00	0 00	0 00	0 00	0 00	100 00	100 00	100 00	100 00	0 00	0 00	0 00	0 00	3 75
Late containment failure %	0 00	100 00	100 00	100 00	100 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	43 47
No containment failure %	100 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 00	0 95

CP&L Response 1.f:

"for each containment release category (including LERF and non-LERF contributors): the associated release frequency, release magnitude (fractions), and MACCS-calculated conditional consequence measures (where available). Please identify those release categories that are considered to contribute to LERF, and those categories to which SGTR and ISLOCA releases are assigned,"

Table 1.f-1 provides a summary of the Level 3 input and output for the RNP SAMA analysis. This table includes the following input information for each release category:

- Frequency (per year)
- RNP Modular Accident Analysis Program (MAAP) case identifier (for reference)
- Airborne release percent at 48 hours for each of the fission product groups provided by MAAP (in this case, Noble Gases, CsI, Te02, SrO, CsOH, and Te2)
- Start time of the airborne release (measured from the time of accident initiation)
- End time of the airborne release (measured from the time of accident initiation)

In addition, the row above the release category identifier indicates whether the release category is defined as a LERF or a non-LERF contributor.

The Level 3 results include the dose-risk (person-rem/yr) and the offsite economic cost-risk before discounting ($/yr). The percentages of each release category composed of SGTR and ISLOCA sequences are also provided. Note that the contributions from these two initiators are completely contained within the LERF release categories 4C, 5, and 5C.

Table 1.f-1: Summary of Level 3 Input and Output

	Non-LERF							LERF					
	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Bin Frequency	2 33E-05	1 08E-05	2.22E-07	5 15E-06	2 62E-07	7.78E-07	3 17E-09	3 74E-08	1.81E-07	0 00E+00	3.70E-06	1 28E-06	3 94E-07
RNP PSA MAAP Run Identifier (RNP PRA, Section 9, "Source Terms and Release Categories," CP&L)	CA-3BA	CA-6B-02	CA-2M-02	CA-10B-01	CA-4SBO-HLF	CA-2B-ISOL3-DCH3	CA-2B-ISOL3-DCH3 INCREASED IN PROPORTION OF RC-2 TO RC-2B FOR NON-VOLATILE (VOLATILES ARE THE SAME AS RC-3)	CA-19E-001	CA-4BSBO-ISO-HLF-01	CA-7B-01	CA-7B-01	CA-7X-01	CA-7X-01
Fission Product Data													
Noble Gases													
Airborne Release % at 48 Hours	0 18	100	100	100	100	20	20	100	91	39	39	92	92
Start of Release (hr)	6	26 5	24 5	35	21 5	3 5	3.5	0 5	4	13	13	6	6
End of Release (hr)	36	26 5	24 5	35	21.5	7	7	15	12	13	13	6	6
CsI													
Airborne Release % at 48 Hours	9 20E-04	0 18	14 8	0 75	4.71	0 08	0 08	2 63	17 1	1.7	1 7	26	26
Start of Release (hr)	6	26 5	24 5	35	21 5	2 5	2 5	0 5	4	13	13	6	6
End of Release (hr)	36	26 5	34	38	21 5	3 5	3 5	0 5	13	13	13	6	6
TeO2													
Airborne Release % at 48 Hours	0	0	0	2.73	2 73	0	0	0	6 2	0	0	0	0
Start of Release (hr)	6	N/A	N/A	36	21 5	N/A	N/A	N/A	12	N/A	N/A	N/A	N/A
End of Release (hr)	36	N/A	N/A	42	31	N/A	N/A	N/A	12	N/A	N/A	N/A	N/A

Table 1.f-1: Summary of Level 3 Input and Output

	Non-LERF							LERF					
	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
SrO													
Airborne Release % at 48 Hours	4 50E-05	7 40E-04	1 00E-04	0 05	0 08	8 60E-04	0 11	0 015	0 39	1 70E-03	1 70E-03	1 10E-01	1.10E-01
Start of Release (hr)	6	26 5	24 5	35	21 5	3 5	3 5	0 5	4	13	13	6	6
End of Release (hr)	36	26 5	24 5	42	21 5	3.5	3 5	0 5	14	13	13	6	6
CsOH													
Airborne Release % at 48 Hours	9 50E-04	0 23	8 74	0 69	5 82	0 05	0 05	2 83	19 6	1 7	1 7	25	25
Start of Release (hr)	6	26 5	24 5	35	21 5	2 5	2 5	0 5	4	13	13	6	6
End of Release (hr)	36	26 5	34	38	21 5	3.5	3 5	0 5	13	13	13	6	6
Te2													
Airborne Release % at 48 Hours	4 00E-07	5 40E-03	2.10E-03	2 16	4 96	0 06	0 06	4 30E-04	3	1 40E-05	1 40E-05	2 30E-04	2.30E-04
Start of Release (hr)	6	26.5	24 5	35	21 5	3 5	3 5	1 2	10 5	14.2	14 2	7 5	7 5
End of Release (hr)	36	26 5	24 5	35	21 5	3.5	3 5	1 2	13	14 2	14 2	12	12
OUTPUT													
Dose -Risk (person-rem/yr)	1 01E-01	2.18E+00	2 95E-01	2 01E+00	1 64E-01	9.73E-02	6 07E-04	2 39E-02	2 79E-01	0 00E+00	1 56E+00	3 04E+00	9 38E-01
Offsite Economic Cost-risk ($/yr)	3 64E+02	9 02E+02	7.24E+02	1 96E+03	4 40E+02	2 80E+01	1 00E+00	4 20E+01	7.22E+02	0 00E+00	3 08E+03	4 35E+03	1.34E+03
OTHER													
Percent of Release Category Composed of SGTR	0	0	0	0	0	0	0	0	0	0	100	0	82 9
Percent of Release Category Composed of ISLOCA	0	0	0	0	0	0	0	0	0	0	0	100	17 1

Table Notes - Puff releases are denoted in the table by those entries with equivalent start and end times
- Only 6 fission product groups are reported in the RNP MAAP results

CP&L Response 1.g:

"justification for neglecting large late release categories in establishing the baseline estimate of offsite consequences, given that large late releases could result in population doses comparable to those for large early releases. Include a justification for not using RC-1A and/or RC-1BA to represent large late releases, given that these release categories result in greater releases of volatile fission products and potentially greater releases of non-volatile fission products than RC-1B,"

The NRC's Severe Accident Policy Statement, the NRC's Safety Goal Policy Statement, and Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," have considered it important to characterize adequate containment performance. PSA techniques have been used by utilities to address the characterization of adequate containment performance. These same techniques have been identified by the PSA Applications Guide (EPRI TR-105396), and by the NRC in Regulatory Guide 1.174, for characterizing containment performance using the LERF parameter for assessing applications using PSA. The PSA Applications Guide states, "Core damage frequency (CDF) is the preferred Level 1 PSA figure of merit. Large Early Release Frequency is the preferred Level 2 figure of merit. In combination, these figures address both prevention (CDF) and mitigation (LERF) and provide assurance that both early and long term health effects are considered." In addition, NUREG/CR-6595 describes LERF as a "suitable metric for making risk-informed regulatory decisions." As LERF has typically been viewed as a Level 2 figure of merit in the industry, the SAMA analysis used only the LERF releases as input for the Level 3 model. Additional work could have been performed to include the contributions of other release categories, but the conclusions of the analysis should not be influenced by this effort. It was also noted that previously accepted SAMA submittals were based on LERF models, and that the use of LERF as the sole input to the Level 3 model was an acceptable approach.

NRC review of the RNP release category fission product magnitudes has raised questions related to the exclusion of large late releases in the Level 3 analysis. Specifically, attention has been drawn to release categories RC-1A and RC-1BA. The rationale for not including these release categories in the Level 3 analysis is presented below.

Review of Table 1.f-1 demonstrates that release categories RC-1A and RC-1BA are comparable to LERF release categories RC-2B, RC-5, and RC-5C in fission product release percentages. Table 1.f-1 also demonstrates that the frequencies for these release categories are the same order of magnitude as RC-2B. Since RC-2B represents such a small contribution to risk, it is inferred that the dose-risk and offsite economic cost-risk for these non-LERF categories would be minimal contributors to the Level 3 results. Given that the Level 3 results (dose-risk and offsite economic cost-risk) contribute only 25% of the total maximum averted cost-risk, inclusion of RC-1A and RC-1BA was not considered to be necessary for the RNP SAMA analysis.

In response to this RAI, work has been performed to quantify the non-LERF contributions to the dose-risk and offsite economic cost-risk. As can be determined from Table 1.f-1, the non-LERF dose-risk and offsite economic cost-risk are 83% and 43% of their LERF counterparts, respectively. These results were incorporated into the evaluation of the RNP maximum averted cost-risk to yield an increase of 14.3% (i.e., $148,033). This is less than the increase shown in both the LERF and real discount rate sensitivity cases that were included as part of the SAMA submittal. Furthermore, it was demonstrated in the real discount rate sensitivity that even a 21% increase in the averted cost-risk calculations would not impact the conclusions of the analysis. Inclusion of the non-LERF release categories in the SAMA cost benefit analysis would not alter the conclusions of the study. The response to NRC Request 6 provides a summary of the Phase 2 cost benefit calculations after inclusion of the non-LERF contributors.

CP&L Response 1.h:

"the definition of LERF used to distinguish a large -early release from a small-early or a large-late release,"

LERF consists of the total frequency of all release classes that occur under the early containment failure or containment bypass categories of the containment failure mode matrix. Note, however, that early small isolation failures have been excluded because they represent accident sequences where the debris has been recovered in vessel, only moderate releases have occurred, and scrubbing is present; or, they are small isolation failures with vessel failure and all containment safeguards functioning so that the debris is heavily scrubbed and there is negligible containment pressure.

CP&L Response 1.i:

"clarification of whether the reported CDF and LERF is per reactor year or per calendar year."

Because RNP's capacity factor in the recent past has been relatively high, there is little difference between calendar year and reactor year as a basis for frequency. The RNP PSA model is quantified to obtain the CDF and LERF assuming that the plant is operating at power. Low power, transition modes, and shutdown risk are not quantified. The calculated frequency is not adjusted based on any assumed plant availability factor or capacity factor.

NRC Request 2:

"It is not clear that the set of SAMAs evaluated in the environmental report (ER) address the major risk contributors for RNP. In this regard, please provide the following:

a. a description of how the dominant risk contributors at RNP, including dominant sequences and cutsets from the PSA and equipment failures and operator actions identified through importance analyses, were used to identify potential plant-specific SAMAs for RNP. Indicate how many sequences and cutsets were considered and what percentage of the total CDF they represent,

b. a listing of equipment failures and human actions that have the greatest potential for reducing risk at RNP based on importance analysis and cutset screening,

c. for each dominant contributor identified in (b), provide a cross-reference to the SAMA(s) evaluated in the ER that address that contributor, and

d. a list of the subset of SAMAs (Table F-8, Phase 1 SAMAs) that are considered unique/specific to Robinson, since it is not clear from the "Source Reference" in the table."

CP&L Response 2.a:

"a description of how the dominant risk contributors at RNP, including dominant sequences and cutsets from the PSA and equipment failures and operator actions identified through importance analyses, were used to identify potential plant-specific SAMAs for RNP. Indicate how many sequences and cutsets were considered and what percentage of the total CDF they represent,"

The most important means of identifying plant specific improvements for the RNP SAMA analysis was a review of the plant's IPE. As part of the IPE, an analysis of RNP's cutsets and importance rankings was performed in order to identify plant weaknesses and to suggest changes that would address the weaknesses identified. In addition to the IPE review, an informal review of the CDF-based and LERF-based Risk Reduction Worth (RRW) rankings for the current model was performed. These rankings were reviewed to determine if any items could be beneficial that were not addressed by the existing SAMA list.

Response 2.b contains additional relevant information, and Response 2.c provides a more detailed discussion of the importance ranking review and the associated results.

CP&L Response 2.b:

"a listing of equipment failures and human actions that have the greatest potential for reducing risk at RNP based on importance analysis and cutset screening,"

The RRW listing has been reviewed down to the 1.033 level. The events with RRW values above 1.033 have been identified in Table 2.b-1. The events below this point would influence the CDF by less than 3.5%. This corresponds to about a $30,000 averted cost-risk based on CDF reduction assuming 100% reliability of the associated event. The events below this point are judged to be unlikely contributors to the identification of cost beneficial enhancements.

The LERF-based RRW factors were also reviewed to determine if there were additional equipment failures or operator actions that should be included in Table 2.b-1. The top contributor not identified in the CDF-based RRW list, OPER-SD (OPERATOR FAILS TO ESTABLISH SHUTDOWN COOLING), corresponded to a benefit of less than $20,000. This benefit includes consideration of the LERF and non-LERF release categories. It should be noted that even if the 1.229 RRW factor for OPER-SD were universally applied to the LERF and non-LERF Level 3 results, which does not account for the fact that the reduction may be distributed through non-contributing release categories, the benefit would be $70,000. Note that this benefit also assumes 100% reliability of OPER-SD, which is conservative. Thus, OPER-SD does not greatly influence the results and is representative of the other equipment and operator failures in the LERF-based RRW list. No events were added to Table 2.b-1 based on the LERF RRW review.

Response 2.c below provides a more detailed discussion of the importance ranking review and relationships of the events to the SAMA list.

Table 2.b-1: RNP Contributors with the Greatest Potential for Reducing Risk				
Number	**Event Name**	**Probability**	**RRW**	**Description**
1	XFL-TQDX	1.00E+00	1.55	SEQUENCE FUNCTIONAL FAILURES INCLUDE TRANSIENT INITIATING EVENT, EVENT Q - LOSS OF REACTOR COOLANT SYSTEM (RCS) INTEGRITY (EARLY), FAILURE OF LONG TERM SD COOLING; FAILURE TO MAINTAIN LONG TERM RCS INVENTORY
2	OPER-4	1.00E+00	1.495	OPERATOR FAILS PROVIDE ALTERNATE COOLING TO CHARGING PUMPS
3	X-OQ-0002	5.00E-03	1.304	RECOVERY VALUE FOR OPER-4 NOT IN COMBINATION WITH OTHER ACTIONS. OPERATOR FAILS PROVIDE ALTERNATE COOLING TO CHARGING PUMPS
4	%T5	3 62E-02	1 301	LOSS OF OFFSITE POWER

Table 2.b-1: RNP Contributors with the Greatest Potential for Reducing Risk				
Number	Event Name	Probability	RRW	Description
5	%T11	1.00E+00	1.257	LOSS OF COMPONENT COOLING WATER
6	~TRANS	1.00E+00	1.196	TRANS. INDUCED LOSS OF DECAY HEAT REMOVAL SEQUENCE MARKER
7	XFL-LQUD	1.00E+00	1.175	SEQUENCE FUNCTIONAL FAILURES INCLUDE: LOSS OF OFFSITE POWER; EVENT Q - LOSS OF RCS INTEGRITY (EARLY); EVENT U - FAILURE OF SAFETY INJECTION (SMALL LOCA); FAILURE OF LONG TERM SD COOLING
8	KCCF%RUN	8 33E-04	1.113	COMMON CAUSE FAILURE OF ALL CCW PUMPS TO RUN
9	X-ACP1	1 56E-01	1.095	LOOP RECOVERY, SEAL LOSS OF COOLANT ACCIDENT (LOCA) AT 1 5 HOURS (ALL START FAILURES)
10	%R	4.96E-03	1.087	STEAM GENERATOR TUBE RUPTURE
11	%T10	3.30E-04	1.083	NON-ISOLABLE SERVICE WATER (SW) PIPE RUPTURE
12	OPER-11	1 00E+00	1.083	OPERATOR FAILS TO IDENTIFY/ISOLATE SW PIPE RUPTURE
13	W%SYSTEM	1.00E+00	1 083	INITIATING EVENTS INVOLVING SW SYSTEM COMPONENTS
14	X-OQ-0010	1.00E-02	1.083	JOINT HUMAN ERROR PROBABILITY FOR OPER-11 (OPERATOR FAILS TO IDENTIFY/ISOLATE SW PIPE RUPTURE) WITH OPER-4 (OPERATOR FAILS PROVIDE ALTERNATE COOLING TO CHARGING PUMPS)
15	%S1	5 30E-03	1.076	SMALL LOCA EVENT
16	~ATWS	1 00E+00	1.075	ATWS SEQUENCE IDENTIFIER
17	XFL-LBU	1 00E+00	1 07	SEQUENCE FUNCTIONAL FAILURES INCLUDE: LOSS OF OFFSITE POWER; EVENT U - FAILURE OF PRIMARY FEED (SAFETY INJECTION); EVENT B - FAILURE OF SECONDARY SIDE HEAT REMOVAL (EARLY)
18	OPER-18B	1.00E+00	1.069	OPERATOR FAILS TO SUPPLY AUXILIARY FEEDWATER (AFW) WITH SW
19	X-ACP3	1.14E-01	1.067	LOOP RECOVERY, SEAL LOCA AT 2.5 HOURS (START FAILURES AND BATTERY DEPLETION)
20	XFL-SDX	1.00E+00	1.067	SEQUENCE FUNCTIONAL FAILURES INCLUDE· SMALL LOCA EVENT, FAILURE TO MAINTAIN LONG TERM RCS INVENTORY; FAILURE OF LONG TERM SD COOLING

Table 2.b-1: RNP Contributors with the Greatest Potential for Reducing Risk				
Number	**Event Name**	**Probability**	**RRW**	**Description**
21	#AMTC	1.50E-01	1.065	PROBABILITY OF MODERATOR TEMPERATURE COEFFICIENT (MTC) BEING LESS NEGATIVE THAN REQUIRED
22	XFL-ATWS9	1.00E+00	1.065	SEQUENCE FUNCTIONAL FAILURES INCLUDE. INITIATING EVENT LEADING TO AN ATWS, REACTOR PROTECTION SYSTEM (RPS) FAILS TO TRIP THE REACTOR; OPERATOR MANUAL REACTOR TRIP (EARLY), FAILURE OF MAIN FEEDWATER; MODERATOR TEMPERATURE COEFFICIENT NOT FAVORABLE
23	#ACBCRDCC	1.00E-05	1.065	COMMON CAUSE FAILURE OF REACTOR TRIP BREAKERS
24	KRV%729NN	4.90E-04	1.063	RELIEF VALVE CC-729 TRANSFER OPEN AND DIVERTS FLOW
25	XFL-TBH	1.00E+00	1.063	SEQUENCE FUNCTIONAL FAILURES INCLUDE TRANSIENT INITIATING EVENTS LEADING TO A LOSS OF DHR; EVENT B - FAILURE OF SECONDARY SIDE HEAT REMOVAL (EARLY); EVENT H - FAILURE TO ESTABLISH PRIMARY BLEED
26	OPER-18A	1.00E+00	1.06	OPERATOR FAILS TO SUPPLY AFW WITH DEEPWELL PUMPS
27	PCCFFOTPLN	1.70E-03	1.055	COMMON MODE FAILURE OF FUEL OIL TRANSFER PUMPS AND VALVES
28	%T9	1 00E+00	1.054	LOSS OF SERVICE WATER
29	XFL-RPX	1 00E+00	1.053	SEQUENCE FUNCTIONAL FAILURES INCLUDE: STEAM GENERATOR TUBE RUPTURE; EVENT X - FAILURE TO ACCOMPLISH COLD LEG RECIRCULATION. EVENT P - SECONDARY-SIDE DEPRESSURIZATION USING SG
30	%T3	9 20E-01	1.053	TURBINE TRIP
31	WCCF%ABCD	1.86E-04	1.049	COMMON CAUSE FAILURE TO RUN ALL SW PUMPS
32	OPER-1	1 00E+00	1.047	OPERATOR FLAG - FAILURE TO SWITCHOVER TO COLD LEG RECIRCULATION
33	XBATBDEP1H	1.00E+00	1.041	BATTERY B DEPLETED AFTER 1 HOUR
34	XBATADEP1H	1.00E+00	1.041	BATTERY A DEPLETED AFTER 1 HOUR
35	OPER-BC	1 00E+00	1.036	OPERATOR FAILS TO CLOSE INPUT BREAKER TO BATTERY CHARGER FOLLOWING UV ON E1/E2
36	RPVCV456FF	2 40E-02	1.034	PORV PCV-456 FAILS TO RECLOSE AFTER DEMAND

Table 2.b-1: RNP Contributors with the Greatest Potential for Reducing Risk				
Number	**Event Name**	**Probability**	**RRW**	**Description**
37	RPVV455CFF	2.40E-02	1.034	PORV PCV-455C FAILS TO RECLOSE AFTER DEMAND
38	KPM%CCWBKR	2.38E-01	1.033	CCW PUMP B FAILS TO RUN FOR A YEAR

CP&L Response 2.c:

"for each dominant contributor identified in (b), provide a cross-reference to the SAMA(s) evaluated in the ER that address that contributor,"

Table 2.c provides a correlation between the events identified in Table 2.b-1 and the SAMAs evaluated in the ER.

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
XFL-TQDX	1.00E+00	1.55	SEQUENCE FUNCTIONAL FAILURES INCLUDE: TRANSIENT INITIATING EVENT; EVENT Q - LOSS OF RCS INTEGRITY (EARLY), FAILURE OF LONG TERM SD COOLING; FAILURE TO MAINTAIN LONG TERM RCS INVENTORY	Sequence marker. It does not provide useful information for SAMA development.
OPER-4	1 00E+00	1.495	OPERATOR FAILS PROVIDE ALTERNATE COOLING TO CHARGING PUMPS	Improvement in operator actions related to support system failures is included in the SAMA list as number 21 A specific evaluation of the benefit of installing self-cooling charging pumps is provided in Response 7.
X-OQ-0002	5.00E-03	1.304	RECOVERY VALUE FOR OPER-4 NOT IN COMBINATION WITH OTHER ACTIONS OPERATOR FAILS PROVIDE ALTERNATE COOLING TO CHARGING PUMPS	Improvement in operator actions related to support system failures is included in the SAMA list as number 21. A specific evaluation of the benefit of installing self-cooling charging pumps is provided in Response 7.

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
%T5	3.62E-02	1.301	LOSS OF OFFSITE POWER	The importance of the LOOP initiator can be addressed through prevention and mitigation. Many SAMAs exist for each of these means and are documented in SAMAs 90-129. Improvement of offsite power availability is the more difficult of the two to address, as a large component of offsite power availability is grid-related However, SAMAs already exist that address offsite power availability (refer to SAMAs 109 and 110). Also, severe weather procedures development is included to address anticipation of a LOOP (SAMA 104) The development of procedures with an emphasis on recovery are also suggested (potentially in switchyard recovery actions) in SAMA 103 No additional SAMAs were suggested for this broad topic.
%T11	1 00E+00	1 257	LOSS OF COMPONENT COOLING WATER	The importance of the Loss of Component Cooling Water (CCW) initiator can be addressed through prevention and mitigation The most important function served by CCW is to support RCP seal cooling (for thermal barrier cooling and seal injection via charging pump cooling). Many SAMAs exist for Improvements Related to Seal LOCAs and are addressed in SAMAs 1-24. No additional SAMAs were suggested for this broad topic.
~TRANS	1.00E+00	1.196	TRANS. INDUCED LOSS OF DHR SEQUENCE MARKER	Sequence marker. It does not provide useful information for SAMA development.

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
XFL-LQUD	1.00E+00	1.175	SEQUENCE FUNCTIONAL FAILURES INCLUDE: LOSS OF OFFSITE POWER; EVENT Q - LOSS OF RCS INTEGRITY (EARLY); EVENT U - FAILURE OF SAFETY INJECTION (SMALL LOCA); FAILURE OF LONG TERM SD COOLING	Sequence marker It does not provide useful information for SAMA development
KCCF%RUN	8.33E-04	1.113	COMMON CAUSE FAILURE OF ALL CCW PUMPS TO RUN	Common Cause Failure is essentially addressed through diversity of systems In general, loss of CCW is treated by SAMAs 1-24, but a specific subset have been identified which address diverse means of providing the major function of CCW (RCP seal cooling). These include SAMAs 11, 12, 15, 19, and 24. No additional SAMAs were suggested.
X-ACP1	1 56E-01	1.095	LOOP RECOVERY, SEAL LOCA AT 1.5 HOURS (All START FAILURES)	LOOP recovery is assumed to be addressed by a specific subset of those SAMAs identified for the LOOP initiator (103, 104, and potentially 109). No additional SAMAs have been identified for improving LOOP recovery.
%R	4.96E-03	1.087	STEAM GENERATOR TUBE RUPTURE	In general, SGTR is treated in the "Improvements in Identifying and Mitigating Containment Bypass" section of the SAMA list (SAMAs 130-152). No additional SAMAs were suggested

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
%T10	3.30E-04	1.083	NON-ISOLABLE SERVICE WATER PIPE RUPTURE	The loss of SW impacts a large number of functions and its severity is exacerbated in this initiator by the fact that it is a non-isolable break. Some SAMAs are included in the list that directly address the loss of SW. SAMA 23 proposes an additional SW pump/pump train to reduce common cause failure, which would include an unisolable break. This would be considered effective if it were an independent train. SAMA 24 suggests an independent seal injection system to reduce the potential for RCP seal damage on the loss of SW. Many other SAMAs indirectly address the loss of SW by proposing alternate means of supporting functions that are normally supplied by SW. These SAMAs include 2, 3, 4, 5, 6, 7, and 13. No additional SAMAs were suggested
OPER-11	1.00E+00	1.083	OPERATOR FAILS TO IDENTIFY/ISOLATE SW PIPE RUPTURE	SAMA 155 addresses improvements in the prevention and mitigation of internal flooding. This is considered to address procedure and training enhancements that may be relevant to this action.
W%SYSTEM	1.00E+00	1.083	INITIATING EVENTS INVOLVING SW SYSTEM COMPONENTS	Sequence marker. It does not provide useful information for SAMA development.

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
X-OQ-0010	1.00E-02	1 083	JOINT HUMAN ERROR PROBABILITY (HEP) FOR OPER-11 (OPERATOR FAILS TO IDENTIFY/ISOLATE SW PIPE RUPTURE) WITH OPER-4 (OPERATOR FAILS PROVIDE ALTERNATE COOLING TO CHARGING PUMPS)	This joint HEP is the failure to align alternate cooling to the charging pumps, given the failure to identify or isolate a SW pipe rupture (loss of SW). Both OPER-11 and OPER-4 are treated above. No additional SAMAs were suggested.
%S1	5 30E-03	1.076	SMALL LOCA EVENT	Many SAMAs are included that address mitigation of small LOCA events These include enhancements to both injection and depressurization methods High pressure make-up is addressed with SAMAs 179, 180, 185, 186, 198, 202, 204, 205, and 209. Depressurization enhancements are addressed in SAMAs 233, 244, and 245. No additional SAMAs were suggested.
~ATWS	1 00E+00	1 075	ATWS SEQUENCE IDENTIFIER	Sequence marker. It does not provide useful information for SAMA development.
XFL-LBU	1.00E+00	1.07	SEQUENCE FUNCTIONAL FAILURES INCLUDE. LOSS OF OFFSITE POWER; EVENT U - FAILURE OF PRIMARY FEED (SAFETY INJECTION), EVENT B - FAILURE OF SECONDARY SIDE HEAT REMOVAL (EARLY)	Sequence marker. It does not provide useful information for SAMA development
OPER-18B	1.00E+00	1.069	OPERATOR FAILS TO SUPPLY AFW WITH SW	This action is directly addressed by the quantification discussed in Response 7 for implementing automatic re-fill of the Condensate Storage Tank (CST). That quantification identifies the benefit of demoting the operator action OPER-18A(B) from a primary action to a back-up action. CST make-up is also addressed in SAMAs 59, 169, and 172

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
X-ACP3	1.14E-01	1.067	LOOP RECOVERY, SEAL LOCA AT 2 5 HOURS (START FAILURES AND BATTERY DEPLETION)	LOOP recovery is assumed to be addressed by a specific subset of those SAMAs identified for the LOOP initiator (103, 104, and potentially 109) No additional SAMAs have been identified for improving LOOP recovery.
XFL-SDX	1.00E+00	1.067	SEQUENCE FUNCTIONAL FAILURES INCLUDE: SMALL LOCA EVENT; FAILURE TO MAINTAIN LONG TERM RCS INVENTORY; FAILURE OF LONG TERM SD COOLING	Sequence marker. It does not provide useful information for SAMA development
#AMTC	1 50E-01	1 065	PROBABILITY OF MTC BEING LESS NEGATIVE THAN REQUIRED	The adequacy of MTC for ATWS mitigation is a function of several variables, the most important of which are considered to be the reliability of AFW, the Pressurizer Power Operated Relief Valves (PORVs) (RCS overpressure protection), Manual Rod Insertion, and the core history RCS overpressure protection has been identified as an issue for ATWS sequences and is addressed by SAMAs 175, 222, and 261. AFW reliability is addressed by multiple SAMAs, including 159, 160, 162, 163, 169, 170, and 173. Rod insertion improvements and other reactivity control schemes are proposed in SAMAs 217, 218, 223, and 228 No additional SAMAs were suggested.
XFL-ATWS9	1.00E+00	1 065	SEQUENCE FUNCTIONAL FAILURES INCLUDE: INITIATING EVENT LEADING TO AN ATWS, RPS FAILS TO TRIP THE REACTOR, OPERATOR MANUAL REACTOR TRIP (EARLY); FAILURE OF MAIN FEEDWATER; MODERATOR TEMPERATURE COEFFICIENT NOT FAVORABLE	Sequence marker It does not provide useful information for SAMA development.

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
#ACBCRDCC	1.00E-05	1.065	COMMON CAUSE FAILURE OF REACTOR TRIP BREAKERS	Additional control over the reactor trip breakers is proposed in SAMA 217. Alternate reactivity control is addressed with SAMA 223. Note that a RRW of 1.065 corresponds to a cost benefit of only about $57,000 based on CDF, and that the Level 2 ATWS impact is limited. This cost benefit range is below credible hardware implementation costs
KRV%729NN	4.90E-04	1.063	RELIEF VALVE CC-729 TRANSFERS OPEN AND DIVERTS FLOW	This item was inadvertently considered to be included in the list as SAMA 16 during the initial review performed for the SAMA analysis in the Environmental Report submittal Charging pump flow diversion is addressed in SAMA 16 and is not representative of CCW flow diversion A new evaluation has been performed as Phase 2 SAMA 10 to evaluate the potential for reducing CCW flow diversion through the relief valve. Results of the cost benefit analysis are provided in Response 6.
XFL-TBH	1 00E+00	1.063	SEQUENCE FUNCTIONAL FAILURES INCLUDE: TRANSIENT INITIATING EVENTS LEADING TO A LOSS OF DHR; EVENT B - FAILURE OF SECONDARY SIDE HEAT REMOVAL (EARLY); EVENT H - FAILURE TO ESTABLISH PRIMARY BLEED	Sequence marker. It does not provide useful information for SAMA development.
OPER-18A	1.00E+00	1.06	OPERATOR FAILS TO SUPPLY AFW WITH DEEPWELL PUMPS	This action is directly addressed by the quantification discussed in Response 7 for implementing automatic re-fill of the CST. That quantification identifies the benefit of demoting the operator action OPER-18A(B) from a primary action to a back-up action. CST make-up is also addressed in SAMAs 59, 169, and 172

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
PCCFFOTPLN	1.70E-03	1.055	COMMON MODE FAILURE OF FUEL OIL TRANSFER PUMPS AND VALVES	This event represents the common cause failure of Emergency Diesel Generators (EDGs) "A" and "B" due to fuel oil transfer pump failure. SAMAs 101 and 105 address this event It should be noted that procedure OP-909 is already in place at RNP, but no credit is taken for it in the PSA model No additional SAMAs were suggested
%T9	1.00E+00	1.054	LOSS OF SERVICE WATER	The loss of SW impacts a large number of functions Several SAMAs are included in the list that directly address loss of SW, as discussed for the SW pipe break initiator (%T10) In addition to those SAMAs, numbers 10 and 20 propose changes that are not necessarily applicable to a SW pipe break scenario. No additional SAMAs were suggested
XFL-RPX	1 00E+00	1.053	SEQUENCE FUNCTIONAL FAILURES INCLUDE: STEAM GENERATOR TUBE RUPTURE, EVENT X - FAILURE TO ACCOMPLISH COLD LEG RECIRC. EVENT P - SECONDARY-SIDE DEPRESSURIZATION USING SG	Sequence marker. It does not provide useful information for SAMA development
%T3	9.20E-01	1.053	TURBINE TRIP	Two SAMAs were identified that would potentially reduce the turbine trip frequency (159, 213). Cost effective means of improving plant availability with respect to operating practices and plant culture are considered to have been addressed through implementation of the Maintenance Rule and PSA applications. No additional, specific SAMAs were suggested for this broad category
WCCF%ABCD	1.86E-04	1.049	CCF TO RUN ALL SW PUMPS	The SAMAs relevant to this event are considered to have been addressed by the %T10 initiator.

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
OPER-1	1.00E+00	1.047	OPERATOR FLAG - FAILURE TO SWITCHOVER TO COLD LEG RECIRCULATION	This SAMA is addressed by the Phase 2 evaluation of automatic switchover to recirculation mode (Phase 2 SAMA 8, Phase 1 SAMA 193).
XBATBDEP1H	1.00E+00	1.041	BATTERY B DEPLETED AFTER 1 HOUR	This event marker is addressed by SAMAs 92 and 96.
XBATADEP1H	1.00E+00	1.041	BATTERY A DEPLETED AFTER 1 HOUR	This event marker is addressed by SAMAs 92 and 96.
OPER-BC	1.00E+00	1.036	OPERATOR FAILS TO CLOSE INPUT BREAKER TO BATTERY CHARGER FOLLOWING UV ON E1/E2	Automatic alignment of the alternate charger could be proposed as a potential change for this case; however, automatic alignment of a charger to a potentially shorted system is not recommended Enhanced training could be proposed, but this is judged to be subsumed by SAMA 128.
RPVCV456FF	2.40E-02	1 034	PORV PCV-456 FAILS TO RECLOSE AFTER DEMAND	This event is considered to be closely related to SAMA 235. SAMA 235 addresses the need to prevent the opening of a PORV in an accident to remove excess energy so that there is no chance of a failure to re-close. It could also be linked to the Boiling Water Reactor (BWR) SAMA for increasing the SRV reseat reliability, although for a BWR the concern is to prevent boron dilution A separate SAMA could be added to specifically address improving reseat reliability after a challenge, but the RRW corresponds to only about $31,000 in averted cost-risk. The LERF-based RRW for this event is only 1 002 and corresponds to a minimal change Level 3 consequences No hardware changes for both relief valves are considered feasible on this cost basis. No new SAMAs are suggested

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
RPVV455CFF	2.40E-02	1.034	PORV PCV-455C FAILS TO RECLOSE AFTER DEMAND	This event is considered to be closely related to SAMA 235. SAMA 235 addresses the need to prevent the opening of a PORV in an accident to remove excess energy so that there is no chance of a failure to re-close. It could also be linked to the BWR SAMA for increasing the SRV reseat reliability, although for a BWR the concern is to prevent boron dilution. A separate SAMA could be added to specifically address improving reseat reliability after a challenge, but the RRW corresponds to only about $31,000 in averted cost-risk. The LERF-based RRW for this event is only 1 002 and corresponds to a minimal change in Level 3 consequences. No hardware changes for both relief valves are considered feasible on this cost basis. No new SAMAs are suggested

Table 2.c: Correlation of Importance Listing to Evaluated SAMAs				
Event Name	**Probability**	**CDF-Based RRW**	**Description**	**Disposition**
KPM%CCWBKR	2.38E-01	1 033	CCW PUMP B FAILS TO RUN FOR A YEAR	This event is related to· 1) the creation of a flow diversion through the check valve of the normally running pump after it fails, and 2) contribution to the loss of CCW through failure of the normally running CCW pump. As discussed in the disposition for KCCF%RUN, several SAMAs exist for mitigating loss of CCW scenarios. Given that the CDF-based RRW for this event corresponds to an averted cost-risk of $30,000 for 100% reliability, and that the LERF-based RRW is 1.0, the possibilities for cost beneficial improvements are limited. Preventative maintenance issues are assumed to be addressed for this risk significant equipment and no large gains are considered to be attainable through further enhancement of maintenance practices. A specific procedural enhancement could be suggested to improve operator response to the flow diversion sequences related to this event, but based on cutset review, flow diversion accounts for an averted cost-risk of only $4,400 No new SAMAs are suggested for this event

CP&L Response 2.d:

"a list of the subset of SAMAs (Table F-8, Phase 1 SAMAs) that are considered unique/specific to Robinson, since it is not clear from the 'Source Reference' in the table."

Several plant specific SAMAs were identified for RNP; however, the industry-based list already included these plant specific SAMAs or other similar SAMAs that addressed the same function. Thus, these items were typically not explicitly included on the RNP SAMA list, as their inclusion did not improve the quality or completeness of the list. The RNP plant specific SAMAs are provided below along with the associated rationale for not creating new SAMA list entries.

A New Procedure for Coping with Flooding Events: This SAMA is intended to assist the operator in identifying flooding sources and potential isolation measures. In addition, the procedures provide steps to limit the accumulation of water and to help prevent equipment damage. SAMAs 155 and 158 address these issues. This item is complete as described in RNP letter dated July 2, 1993. Abnormal Operating Procedure (AOP) AOP-08, "Accidental Release of Liquid Waste," AOP-14, "Component Cooling Water System Malfunction," and AOP-22, "Loss of Service Water," have been revised and a new procedure, AOP-32, "Response to Flooding from the Fire Protection System," was written.

Operation of the Steam-Driven Auxiliary Feedwater Pump in the Self-Cooling Mode: This SAMA reduces the failure probability of the AFW pumps by eliminating the operator action to align self-cooling when normal cooling is lost. This change is considered to be approximately the same as SAMA 165. The "Result of Potential Enhancement" column suggests making the pump self-cooled as an alternative. A new entry was not created for this SAMA. The steam-driven AFW pump was permanently aligned for self-cooling mode as part of a plant modification.

Modification of the Plant Safety-Related Batteries: The IPE identified this change to upgrade the capacity of the safety-related batteries from 1 hour to 4 hours to allow more time for offsite power recovery. This type of change was already included in the SAMA list as number 92 and no new entry was created. Updated results from the RNP PSA model demonstrated that a modification to the station safety-related batteries was not cost beneficial relative to the reduction in core damage frequency that would have been obtained. The SAMA analysis also concluded that this change would not be cost beneficial due to the prohibitive cost of batteries. Additionally, two new procedures were written to cope with a loss of DC power, EPP-26, "Loss of DC Bus A," and EPP-27, "Loss of DC Bus B."

As part of Response 7, a calculation has been performed to quantify the benefits associated with improving the plant's DC capability.

Development of a More Extensive Preventative Maintenance Program for the Dedicated Shutdown Diesel Generator: This change was not included on the SAMA list because this early 1990s era insight is considered to be encompassed by the implementation of the Maintenance Rule. This item is complete as discussed in RNP letter dated July 2, 1993. The preventative maintenance program for the dedicated shutdown diesel generator was revised to be similar to the preventative maintenance program for the emergency diesel generators.

Revision of Safety Injection (SI) and Containment Vessel (CV) Spray System Valve Test Procedure: This item was intended to reduce the ISLOCA frequency by changing the order of valve testing. The RWST isolation valves are to be tested before the SI pump test, so that any open pathway will have a greater probability of being identified. This was not added to the list since an ISLOCA valve test procedure change was already included on the list, i.e., SAMA 143. This item is complete as discussed in RNP letter dated July 2, 1993. Procedure Operations Surveillance Test (OST)-157 has been deleted, and OST-703, "ISI Primary Side Valve Test," was written to replace it.

Test of the HVAC Requirements for the E1/E2 Bus Room: The evaluation of the Heating, Ventilation, and Air Conditioning (HVAC) system was completed to verify the requirements for room cooling used in the IPE. This was not included as a SAMA since it was not a plant change, but an analysis to support PSA assumptions. This item is complete as discussed in RNP letter dated August 12, 1994. A best estimate analysis indicated that HVAC was not required for the E1/E2 room during a severe accident.

Induced Steam Generator Tube Rupture: This change was described as the elimination of the use of the reactor coolant pumps as a last attempt to cool the core. This change involved a significant reduction in the probability of an induced SG tube rupture during a severe accident. This item was included in the implementation of the Severe Accident Management Guidelines (SAMGs) by the WOG. As SAMG implementation was already included in the SAMA list as number 63, no additional entry was made for this change. It has been accounted for in the revision of Function Restoration Procedure (FRP)-C.1, "Response to Inadequate Core Cooling."

Walk-Through of the Long-Term Emergency Core Cooling System Recirculation Procedure: This walk-through was completed to determine whether the human reliability analysis appropriately credited all features of the procedure. This item was not included on the SAMA list since it was performed to support the PSA analysis and improve its accuracy rather than to directly reduce plant risk. This item is complete as discussed in RNP letter dated August 12, 1994. Enhancements were made to End Path Procedure (EPP)-9, "Transfer to Cold Leg Recirculation."

Charging Pumps Self-Cooling Modification: This item would remove the charging pump cooling dependence on CCW by installing new, self-cooled pumps or modifying the current pumps to be self-cooled. Updated results from the RNP PSA model demonstrated that a modification for self-cooling of the charging pumps was not cost beneficial relative to the reduction in core damage frequency that would have been obtained. Other, more cost beneficial means of providing pump cooling were proposed, such as proceduralizing cross-connection to the fire protection system. This procedural change was implemented. In light of the cost of new or modified pumps and the previous cost-benefit analysis, no new SAMA was added to the list for this item. Specific SAMAs already existed in the list that addressed improving RCP seal cooling through hardware and procedural changes. SAMA 4 addresses improving operator response to a loss of CCW and the postulated subsequent loss of charging pumps/RCP seal cooling; SAMAs 9 and 14 deal with removing the RCP seal dependence for cooling; SAMAs 11, 12, 19, and 24 suggest installation of alternate RCP seal cooling systems/methods; and, several SAMAs are included that address preventing a loss of CCW. This item was cancelled as discussed in RNP letter dated August 12, 1994.

As part of Response 7, a calculation has been performed to quantify the benefits associated with the change to incorporate self-cooled charging pumps.

Automatic Re-Fill of the Condensate Storage Tank (CST): This modification addresses the dependence on operator action for scenarios requiring long term availability of the CST. RNP can supply the AFW suction from SW and the diesel fire pump, and the time available for the operator action to align these sources is typically long. As a result, the reliability of these actions is relatively high and only a small benefit relative to the cost of a hardware change would be associated with automating the CST re-fill process. No new SAMA was added to the list, as similar SAMAs exist that address CST make-up and AFW supply (i.e., SAMAs 59, 169, and 172). In addition, the IPE-related evaluation of this project determined that the reduction in core damage frequency resulting from this modification did not justify the cost.

As part of Response 7, a calculation has been performed to quantify the benefits associated with automating the CST re-fill process.

NRC Request 3:

"The SAMA analysis did not include an assessment of SAMAs for external events. The RNP IPEEE study has shown that the CDF due to internal fire initiated events is about 9.2×10^{-5} per reactor year. In addition, the risk analyses at other commercial nuclear power plants indicate that external events could be large contributors to CDF and the overall risk to the public. In this regard, the following additional information is needed:

a. NUREG-1742 ("Perspectives Gained From Individual Plant Examination of External Events (IPEEE) Program," Final Report, 4/02), lists the significant fire area CDFs for Robinson (page 3-26 of Volume 2). While we recognize that these CDFs are often conservative, they are still large in comparison to the Robinson internal events CDF. For each fire area, please explain what measures were taken to further reduce risk and explain why these CDFs cannot be further reduced in a cost-effective manner.

b. NUREG-1742 lists seismic outliers and improvements for Robinson (page 2-30 of Volume 2). Please summarize the disposition of the 33 issues/anomalies related to seismic interactions, maintenance, or housekeeping and the 47 components that were identified as outliers. If no plant modifications were implemented, please explain why within the context of this SAMA study."

CP&L Response 3.a:

". . .For each fire area, please explain what measures were taken to further reduce risk and explain why these CDFs cannot be further reduced in a cost-effective manner."

As part of the IPEEE process, the fire areas with higher CDF results were reviewed for possible modifications or other changes to reduce risk. Procedure changes and modifications were made to reduce CDF in three fire areas associated with Control Room fires, DC cabinet fires, and yard transformer fires. The reduced CDF numbers for these three areas were submitted to NRC by letter dated November 30, 1995, and are therefore already included in NUREG-1742. It should be noted that the results of the IPEEE are not directly comparable with those of the IPE; further actions to reduce risk were not considered warranted. This is discussed further in the following paragraphs.

The methodology used to perform the IPE was based on a systems analysis approach that has achieved an accepted degree of maturity. The analysis of external initiating events, by contrast, has not reached the same degree of maturity. For example, some of the potentially damaging external initiating events have very low frequencies that cannot be estimated using actuarial data without considerable extrapolation, so the frequency estimates are subject to a large uncertainty. Many of the events can occur with a range of severity, with the damage potential being a function of that severity. Because of this, the methods that have been developed to analyze the impact of external initiating events are essentially screening analyses, designed either to identify the most significant contributors while minimizing the need for detailed analysis, or to identify specific weaknesses without explicitly estimating risk. The method chosen to analyze the impact of seismic events, the Seismic Margin method, is the latter type of analysis. There is no estimation of core damage frequency. Instead, the analysis is an assessment of whether the plant has sufficient margin over and above the design basis to withstand the Review Level Earthquake (RLE). The analysis of the Other External Events for RNP is, for the most part, a confirmation that the plant, even though not built to the requirements of the Standard Review Plan criteria, does, in fact, comply with their intent, and does not require that core damage frequency be calculated.

The PSA approach adopted for the fire analysis results in the evaluation of the core damage frequencies from a set of fire scenarios. However, even in this case, the core damage frequency is not evaluated in the same way as for internal initiating events. The analysis is based on a screening approach, in which the fire areas were screened from further consideration when a conservative analysis showed that the frequency of core damage was less than 1.0E-06. However, since for areas that are screened the analysis is not further refined, the degree of conservatism is not estimated. Therefore, it would be inaccurate to sum the screening core damage frequencies to obtain the overall core damage frequency. Instead, the analysis has been used to identify the scenarios that have the highest likelihood of leading to core damage.

Additionally, the sequences in the IPE were grouped by functional type for screening and for comparison with the Severe Accident Issue Closure Guidelines (NUMARC, 1992). In the fire analysis, sequences were grouped by fire location, because it is the vulnerable locations that are of interest.

CP&L Response 3.b:

". . . Please summarize the disposition of the 33 issues/anomalies related to seismic interactions, maintenance, or housekeeping and the 47 components that were identified as outliers. If no plant modifications were implemented, please explain why within the context of this SAMA study."

The disposition of issues identified as a result of efforts related to Unresolved Safety Issue A-46 are discussed in letters from CP&L to the NRC dated November 30, 1995, and August 10, 1998.

NRC Request 4:

"The SAMA analysis did not include an assessment of the impact that PSA uncertainties and external event risk considerations would have on the conclusions of the study. Some license renewal applicants have opted to double the estimated benefits (for internal events) to accommodate any contributions for other initiators when sound reasons exist to support such a numerical adjustment, and to incorporate additional margin in the SAMA screening criteria to address uncertainties in other parts of the analysis (e.g., an additional factor of two in comparing costs and benefits of each SAMA). Please provide the following information to address these concerns:

a. an estimate of the uncertainties associated with the calculated core damage frequency (e.g., the mean and median CDF estimates and the 5th and 95th percentile values of the uncertainty distribution),

b. an assessment of the impact on the Phase 1 screening if risk reduction estimates are increased to account for uncertainties in the risk assessment and the additional benefits associated with external events, and

c. an assessment of the impact on the Phase 2 evaluation if risk reduction estimates are increased to account for uncertainties in the risk assessment and the additional benefits associated with external events. Please consider the uncertainties due to both the averted cost-risk and the cost of implementation to determine changes in the net value estimate for these SAMAs. (Note that some of the SAMA candidates; e.g., Phase II SAMA 3 and 7 could potentially become cost-beneficial. Also, note that the cost for Phase II SAMA 3 is given as $50K in Table F-9 and as >$280K in Section F.6.3. Please clarify.)"

CP&L Response 4.a:

"an estimate of the uncertainties associated with the calculated core damage frequency (e.g., the mean and median CDF estimates and the 5th and 95th percentile values of the uncertainty distribution),"

An estimate of the uncertainty inherent in the RNP Level 1 PSA model has been calculated using the UNCERT code and is summarized as follows:

Parameter	Value
Mean CDF	4.54 E-05
Median CDF	3.32 E-05
5th Percentile of Uncertainty Distribution	1.51 E-05
95th Percentile of Uncertainty Distribution	1.06 E-04

CP&L Response 4.b:

"an assessment of the impact on the Phase 1 screening if risk reduction estimates are increased to account for uncertainties in the risk assessment and the additional benefits associated with external events,"

The results of the Phase 1 screening process can be impacted by incorporating external event contributions or implementing conservative values from the PSA uncertainty distribution. Inclusion of external events or use of the 95th percentile PSA results will increase the maximum averted cost-risk and prevent the screening of some higher cost modifications. However, the impact on the overall SAMA results due to the retention of the higher cost SAMAs for Phase 2 analysis is small. The benefit from the implementation of those SAMAs must be large in order to be cost beneficial. The changes associated with the Phase 2 analysis are discussed in Response 4.c.

The impact of uncertainty in the PSA results and the consequences of including external events contributions in the Phase 1 SAMA analysis have been examined. The maximum averted cost-risk is the primary Phase 1 criteria based on the effect of PSA uncertainty or inclusion of external events contributions. Thus, this response focused on recalculating the maximum averted cost-risk given consideration of these factors, and re-performing the Phase 1 screening process. Other factors, such as estimated costs of implementation, can impact the Phase 1 results. However, these cost estimates are generally considered to be conservatively low estimates and use of these estimates will more likely result in the retention of SAMAs that are not cost beneficial than in the screening of potentially important SAMAs.

As discussed in Response 3, the methods and technology available to perform the external events evaluation at RNP have not reached the same level of maturity as those implemented in the internal events analysis. The external events analysis is primarily a screening study used to identify weaknesses based on relative risk. For areas where large uncertainties exist in event probabilities, conservative estimates are employed. The end result is one that will likely identify important components or scenarios for a given plant, but will not provide a core damage frequency that can be compared to one that has been developed for internal events. While the RNP external events analysis evaluated plant strengths and weaknesses, the available core damage frequencies reported in the IPEEE and its amendments are not appropriate for use in this RAI response. As a bounding estimate, external events are considered to contribute an amount equal to the internal events. Thus, the "baseline case" would be modified to develop a revised maximum averted cost-risk based on a factor of 2 increase in the CDF and Level 3 results. This revision would result in a CDF of 8.64E-5/yr, a dose-risk of 21.4 person-rem/yr, and an offsite economic cost-risk of \$27,168/yr. The corresponding maximum averted cost-risk is \$2.36 million.

Use of the 95th percentile PSA results yields a slightly larger result than the factor of two multiplier implemented to account for external events. As such, a review of the Phase 1 analysis using the 95th percentile PSA results is considered to bound the external events case.

The PSA uncertainty calculation results, which are presented in Response 4.a, identify the 95th percentile CDF as 1.06E-4/yr. This is a factor of 2.45 greater than the CDF point estimate produced by the RNP PSA. As the same type of uncertainty analysis was not available for the Level 2 and Level 3 results, the 95th percentile Level 3 results were estimated. The dose-risk and offsite economic cost-risk were increased by a factor of 2.45 to simulate the increase in the CDF resulting from the use of the 95th percentile results. The 95th percentile dose-risk and offsite economic cost-risk are 26.2 person-rem/yr and \$33,281/yr, respectively. The corresponding maximum averted cost-risk is \$2.89 million.

The initial SAMA list has been re-examined using the revised maximum averted cost-risk to identify SAMAs that would be retained for the Phase 2 analysis. Those SAMAs that were previously screened due to costs of implementation that exceeded \$1,033,000 are now retained if the costs of implementation are less than \$2.89 million. Table 4.b-1 identifies the additional SAMAs that would be passed to the Phase 2 analysis given the use of the 95th percentile PSA results.

Since the changes made to account for external events are based on conservative estimates, use of the 95th percentile PSA results in conjunction with the external events contributions produces overly conservative results and reduces the usefulness of the analysis. The combined effects of including external events and the 95th percentile PSA results in the SAMA analysis would allow several additional high cost SAMAs to reach Phase 2. However, given that the 95th percentile results were used to allow them to pass the Phase 1 screening process, these events are not considered cost beneficial.

Table 4.b-1: Additional SAMAs Retained for Phase 2 Analysis Given Inclusion of External Events

Phase I SAMA ID Number	SAMA Title	Result of Potential Enhancement	Disposition Given Inclusion of External Events
19	Use fire protection system pumps as a backup seal injection and high-pressure makeup.	SAMA would reduce the frequency of the RCP seal LOCA and the SBO CDF.	Fire Protection (FP) is a low head system at RNP and cannot be used as a high pressure injection source Modifications to convert it to a high pressure system are estimated to be more than the cost estimated for installing a new high pressure system The cost of installing a new and separate passive high pressure system has been conservatively estimated at approximately $1.7 million. However, as this estimate is less than the RNP 95th percentile maximum averted cost-risk of $2.89 million, it has been identified as a SAMA that would be retained for Phase 2 evaluation.
57	Provide a reactor vessel exterior cooling system.	SAMA would provide the potential to cool a molten core before it causes vessel failure, if the lower head could be submerged in water.	The cost of this enhancement has been estimated to be $2.5 million This is less than the RNP 95th percentile maximum averted cost-risk of $2.89 million, and has been identified as a SAMA that would be retained for Phase 2 analysis
92	Provide additional DC battery capacity.	SAMA would ensure longer battery capability during a SBO, reducing the frequency of long-term SBO sequences.	The cost of implementation for this SAMA has been estimated to be greater than $5 million This is greater than the maximum averted cost-risk based on the 95th percentile PSA results ($2.89 million) and has been identified as a SAMA that would not be retained for Phase 2 analysis. However, more cost beneficial means of improving plant DC capability have been identified. These means are considered in the Phase 2 disposition that is described in Response 4 c

Table 4.b-1: Additional SAMAs Retained for Phase 2 Analysis Given Inclusion of External Events

Phase I SAMA ID Number	SAMA Title	Result of Potential Enhancement	Disposition Given Inclusion of External Events
93	Use fuel cells instead of lead-acid batteries	SAMA would extend DC power availability in an SBO.	The cost of implementation for this SAMA has been estimated to be $2 million This is less than the RNP 95th percentile maximum averted cost-risk of $2 89 million, and has been identified as a SAMA that would be retained for Phase 2 analysis. In addition, more cost beneficial means of improving plant DC capability have been identified These means are considered in the Phase 2 disposition that is described in Response 4 c.
100	Create AC power cross-tie capability with other unit	SAMA would improve AC power reliability.	The estimated cost of implementation for SAMA 123 has been used as a lower bound estimate for this SAMA ($1.2 million) This is less than the RNP 95th percentile maximum averted cost-risk of $2.89 million, and has been identified as a SAMA that would be retained for Phase 2 analysis.
139	Install additional instrumentation for ISLOCAs.	SAMA would decrease ISLOCA frequency by installing pressure or leak monitoring instruments in between the first two pressure isolation valves on low-pressure injection lines, RHR suction lines, and HPSI lines	The cost of implementation for this SAMA has been estimated at $2 3 million This is less than the RNP 95th percentile maximum averted cost-risk of $2.89 million, and has been identified as a SAMA that would be retained for Phase 2 analysis.
164	Install a new condensate storage tank (CST)	Either replace the existing tank with a larger one, or install a back-up tank	While the $1 million estimate for this SAMA's cost of implementation is considered to be a conservatively low estimate, no other estimate has been developed that is greater than the RNP 95th percentile maximum averted cost-risk of $2.89 million This SAMA has been identified as one that would be retained for Phase 2 analysis.

Table 4.b-1: Additional SAMAs Retained for Phase 2 Analysis Given Inclusion of External Events

Phase I SAMA ID Number	SAMA Title	Result of Potential Enhancement	Disposition Given Inclusion of External Events
178	Provide the capability for diesel driven, low pressure vessel make-up.	This SAMA would provide an extra water source in sequences in which the reactor is depressurized and all other injection is unavailable (e.g., FP system).	Based on engineering judgment and similarities to SAMA 179, the installation of a new, diesel driven, low pressure injection system was judged to exceed the maximum averted cost-risk in the submittal. As a more exact estimate was not prepared, this SAMA has been identified as one that would be retained for Phase 2 analysis, given the use of the 95th percentile PSA results and non-LERF contributions to the maximum averted cost-risk.
185	Upgrade Chemical and Volume Control System to mitigate small LOCAs.	For a plant where the Chemical and Volume Control System cannot mitigate a Small LOCA, an upgrade would decrease the Small LOCA CDF contribution	The cost of implementation for this modification is based on the cost estimated to install a passive high pressure injection system ($1.7 million). This was considered to be a conservatively low estimate for an active high pressure system, but similar in scope to the changes required to upgrade the current Chemical and Volume Control System (CVCS). This estimate is less than the RNP 95th percentile maximum averted cost-risk of $2.89 million, and has been identified as a SAMA that would be retained for Phase 2 analysis.
202	Passive High Pressure System	SAMA will improve prevention of core melt sequences by providing additional high pressure capability to remove decay heat through an isolation condenser type system	The cost of this enhancement has been estimated to be $1.7 million This is less than the RNP 95th percentile maximum averted cost-risk of $2.89 million, and has been identified as a SAMA that would be retained for Phase 2 analysis.

Table 4.b-1: Additional SAMAs Retained for Phase 2 Analysis Given Inclusion of External Events			
Phase I SAMA ID Number	**SAMA Title**	**Result of Potential Enhancement**	**Disposition Given Inclusion of External Events**
233	Create/enhance RCS depressurization ability.	With either a new depressurization system, or with existing PORVs, head vents, and secondary side valve, RCS depressurization would allow earlier low pressure Emergency Core Cooling System (ECCS) injection Even if core damage occurs, low RCS pressure would alleviate some concerns about core melt ejection.	The cost of implementation for this SAMA has been estimated to range between $500,000 and $4 6 million. While it is expected that the cost of implementation would be closer to $4 6 million than to $500,000, this SAMA has been identified as one that would be retained for Phase 2 analysis.

CP&L Response 4.c:

"an assessment of the impact on the Phase 2 evaluation if risk reduction estimates are increased to account for uncertainties in the risk assessment and the additional benefits associated with external events. Please consider the uncertainties due to both the averted cost-risk and the cost of implementation to determine changes in the net value estimate for these SAMAs. (Note that some of the SAMA candidates; e.g., Phase II SAMA 3 and 7 could potentially become cost-beneficial. Also, note that the cost for Phase II SAMA 3 is given as $50K in Table F-9 and as >$280K in Section F.6.3. Please clarify.)"

As discussed in Response 4.b above, the 95th percentile PSA results are more limiting than the factor of two increase applied to the results to account for external events contributions. Thus, no specific case is examined to identify the impact of including the external events contributions, since those effects are bounded by the 95th percentile PSA results case.

In order to perform this assessment, it was necessary to make an assumption about the 95th percentile PSA results for the Level 2 and 3 analyses. This is due to the fact that the same type of uncertainty analysis that was performed as part of Response 4.a is not available for the Level 2 and 3 models. The assumption that has been made is that the 95th percentile results for the Level 2 and 3 models can be represented by increasing the base dose-risk and offsite economic cost-risk in proportion to the Level 1 results.

The PSA uncertainty calculation, which is presented in Response 4.a, identifies the 95th percentile CDF as 1.06E-4/yr. This is a factor of 2.45 greater than the CDF point estimate produced by the RNP PSA. As discussed in Response 4.b, the 95th percentile dose-risk and offsite economic cost-risk are 26.2 person-rem/yr and $33,281/yr, respectively. The corresponding maximum averted cost-risk is $2.89 million. The factor of 2.45 is also assumed to propagate through the results for the model runs performed for the Phase 2 detailed calculations. This means that the averted cost-risk for each case will be increased by the same factor.

Table 4.c-1 summarizes the results of the Phase 2 dispositions for additional SAMAs retained using the 95th percentile PSA results. The SAMAs with costs of implementation between $1,033,000 (i.e., maximum averted cost-risk from the ER submittal) and $2,890,000 (i.e., maximum averted cost-risk using the 95th percentile PSA results) that were previously screened on high cost are now considered further. None of the new Phase 2 SAMAs were judged to be potentially cost beneficial, and it was not considered necessary to perform detailed model quantifications to demonstrate this. PSA insights and the results of other, similar model changes were used to assess the potential benefits of these high cost modifications.

Table 4.c-2 provides a summary of the impact of using the 95th percentile PSA results in the detailed cost benefit calculations that were performed for the ER submittal. In addition, the new plant specific SAMA identified in Response 2.c is addressed as Phase 2 SAMA number 10. The initial results indicate that Phase 2 SAMAs 3 and 7 are cost beneficial when the 95th percentile PSA results are used. However, review of the assumptions used in the ER submittal to estimate the impact of these SAMAs shows them to be overly optimistic. When a more appropriate assessment of the risk reduction offered by these SAMAs is applied, the associated averted cost-risk decreases, and the SAMAs are no longer cost beneficial. These two SAMAs are discussed in more detail below.

<u>Phase 2 SAMA 3, Increase Frequency for Valve Leak Testing</u>: The averted cost-risk for this SAMA is based on eliminating all risk modeled for the ISLOCA initiating event. Increased testing may in fact increase the ISLOCA frequency rather than decreasing it. This is due to the additional valve manipulations in the ISLOCA pathways and the added probability that one of the valves may become mispositioned. Even if this consideration is discounted, the reduction in ISLOCA frequency would be less than the complete prevention of the accident category.

A 20% reduction in the risk associated with the ISLOCA (CDF=3%) is considered optimistic for this SAMA, but if such a reduction were applied, the corresponding averted cost-risk is less than $69,000.

It should also be noted that the cost of implementation used for this SAMA is only based on one day of replacement power. This was used to indicate that more frequent ISLOCA testing would require a plant shut down to allow access to the valves inside the containment biological shield wall. No consideration was given to the costs of revising procedures or the manpower needed to perform this testing.

Therefore, even using the 95th percentile PSA results, this SAMA is not cost beneficial.

Phase 2 SAMA 7, Implement RWST Make-Up Procedure: This SAMA has two potentially different applications for RNP. The first is a procedure enhancement, and the second is a procedure enhancement coupled with a hardware modification to increase make-up capacity.

A procedure currently exists that directs re-fill of the RWST; however, this is a normal or routine operational procedure that is not integrated into the emergency operating procedure structure. The current PSA model credits use of the normal procedure for late RWST re-fill for SGTR scenarios. A potential enhancement would be to incorporate the use of this procedure into the Emergency Operating Procedures (EOPs), as discussed in the ER submittal. However, the ER submittal assumed that this enhancement would result in 100% reliability of this action, which is overly optimistic. Given the availability of the Emergency Response Organization during the long time frame where this action would be applicable, the benefit of including a reference to the RWST make-up procedure would be limited, since a high degree of confidence exists that this mitigation strategy would be employed regardless of any procedural linkage to the EOPs. Even if this procedure modification could reduce the failure rate of the RWST re-fill action by 50%, which is also considered optimistic, the averted cost-risk for this SAMA is $40,000. This is less than the $50,000 estimate for researching, writing, implementing, and training operators on a new procedure. In addition, the $40,000 averted cost-risk is based on the 95th percentile PSA results, which are conservative. Therefore, this SAMA is not cost beneficial.

The other option for this SAMA would be to include a hardware modification in addition to the procedural change. The hardware modification would be required to increase the make-up flowrate so that the system could be used in Small Break Loss of Coolant Accident (SBLOCA) or ISLOCA scenarios. This change would increase the benefit of the make-up system, but the hardware change would be costly. If all risk from SBLOCA (see Table 4.c-1, Phase 1 SAMA 185) and ISLOCA (see Table 4.c-1, Phase 1 SAMA 139) are eliminated, and the $40,000 averted cost-risk for SGTR sequences is considered, the averted cost-risk is $589,000. This averted cost-risk assumes complete elimination of the ISLOCA and SBLOCA initiating events, while the actual benefit would only be a fraction of this estimate. The cost of larger pumps, greater capacity boration equipment, larger piping, and new power sources would exceed the potential averted cost-risk.

Summary: The use of the 95th percentile PSA results (or including external events contributions) does not impact the results of the SAMA analysis. New, high cost SAMAs were retained from the Phase 1 analysis as a result of the higher maximum averted cost-risk, but these high cost items had relatively low averted cost-risks associated with their implementation. None were identified as potentially cost beneficial. Those SAMAs that were analyzed in the Phase 2 analysis in the ER were re-examined. Use of the 95th percentile PSA results in conjunction with the original estimates of the SAMAs' impacts on the model resulted in the classification of two SAMAs as potentially cost beneficial (Phase 2 SAMAs 3 and 7). However, SAMAs 3 and 7 were ultimately shown not to be cost beneficial when more appropriate estimates of their benefits were used.

Table 4.c-1: Phase 2 Dispositions for Additional SAMAs Retained from the Phase 1 Analysis Given Use of the 95th Percentile PSA Results

Phase 1 SAMA ID Number	SAMA Title	Result of Potential Enhancement	Cost of Implementation	Phase 2 Disposition Given the Use of the 95th Percentile PSA Results
19	Use fire protection system pumps as a backup seal injection and high-pressure makeup.	SAMA would reduce the frequency of the RCP seal LOCA and the SBO CDF.	$1.7 million	The $1.7 million cost of implementation for this SAMA is more than half of the $2 89 million cost-risk for the 95th percentile case The benefits associated with the improvements to the RCP seal cooling system that are suggested in this SAMA should be consistent with the benefit gained from removing the cooling dependence for the charging pumps. This was examined in Response 7 and the averted cost-risk was determined to be about $336,000. For the 95th percentile case, the 2 45 scaling factor is applied to yield a benefit of $823,000 The largest RRW associated with this function that has been identified is the common cause failure of the suction valves from the RHR system (1 006) Common cause failure of the HHSI pumps is even lower, at 1.004. Thus, the additional benefit from improving the high pressure injection function is small and would not offset the additional $800,000 required to make this SAMA cost beneficial
57	Provide a reactor vessel exterior cooling system	SAMA would provide the potential to cool a molten core before it causes vessel failure, if the lower head could be submerged in water.	$2 5 million	This SAMA only impacts post-core damage accident mitigation; it does not play a part in accident prevention If all Level 3 results (dose-risk and offsite economic cost-risk) are assumed to be decreased to zero based on this modification, the averted cost-risk is only $922,176, which is less than the cost of implementation This SAMA is not cost beneficial and is screened from further analysis.
92	Provide additional DC battery capacity.	SAMA would ensure longer battery capability during an SBO, reducing the frequency of long-term SBO sequences	RNP-specific analysis has shown the cost of installing new batteries exceeds $5,000,000. In addition, portable DC chargers were shown to cost in excess of $350,000 per year due to the need to increase the size of the operations staff	The averted cost-risk associated with implementing improved DC power capability has been analyzed in Response 7. It was determined that the averted cost-risk for this enhancement is $47,000 based on the PSA point estimate results If a factor of 2 45 is applied to the averted cost-risk to account for use of the 95th percentile PSA results, the averted cost-risk becomes $115,150 This is less than the cost of new batteries or for adding portable DC charging capabilities

Table 4.c-1: Phase 2 Dispositions for Additional SAMAs Retained from the Phase 1 Analysis Given Use of the 95th Percentile PSA Results

Phase 1 SAMA ID Number	SAMA Title	Result of Potential Enhancement	Cost of Implementation	Phase 2 Disposition Given the Use of the 95th Percentile PSA Results
93	Use fuel cells instead of lead-acid batteries.	SAMA would extend DC power availability in an SBO	$2 million	The averted cost-risk associated with implementing improved DC power capability has been analyzed in Response 7. It was determined that the averted cost-risk for this enhancement is $47,000 based on the PSA point estimate results. If a factor of 2 45 is applied to the averted cost-risk to account for use of the 95th percentile PSA results, the averted cost-risk becomes $115,150 This is less than the $2 million estimate for installation of fuel cells.
100	Create AC power cross-tie capability with other unit	SAMA would improve AC power reliability.	$1.2 million	The RRW of LOOP initiator is 1.3 based on CDF, and 1.02 based on LERF. Even if a conservative assumption is made that installing a cross-tie to the Darlington unit would be equivalent to reducing the LOOP contribution to zero, the reduction in the risk would only be a factor of 1.3 This corresponds to an averted cost-risk of about $667,000, which is less than the $1 2 million estimated cost of implementation
139	Install additional instrumentation for ISLOCAs	SAMA would decrease ISLOCA frequency by installing pressure or leak monitoring instruments in between the first two pressure isolation valves on low-pressure injection lines, RHR suction lines, and HPSI lines	$2.3 million	It was shown in the SAMA submittal that elimination of all ISLOCA risk resulted in an averted cost-risk of $140,455 (Phase 2 SAMA number 3). Response 6 provides an averted cost-risk for this same SAMA after accounting for the non-LERF contributions, which only increased the estimate to $140,778 If the 95th percentile scaling factor of 2 45 is applied to this result, the averted cost-risk becomes $344,906. This is less than the estimated cost of implementation
164	Install a new condensate storage tank (CST)	Either replace the existing tank with a larger one, or install a back-up tank.	$1 million for a new CST, $484,000 for a connection to the Service Water System	This SAMA addresses the long term availability of the CST Response 7 provides an averted cost-risk for installing a vacuum breaker between the CST and the Service Water System, which is approximately the functional equivalent of increasing the size of the CST. The cost of the modification is also less expensive than providing a new, larger CST. The averted cost-risk for automatic alignment of an alternate AFW suction source is estimated to be $75,305 If the 95th percentile scaling factor of 2 45 is applied to this result, the averted cost-risk becomes about $184,500 This is less than the cost of a new CST. The cost of installing a connection to the Service Water System that would re-fill the CST yields a negative net value.

Table 4.c-1: Phase 2 Dispositions for Additional SAMAs Retained from the Phase 1 Analysis Given Use of the 95th Percentile PSA Results				
Phase 1 SAMA ID Number	**SAMA Title**	**Result of Potential Enhancement**	**Cost of Implementation**	**Phase 2 Disposition Given the Use of the 95th Percentile PSA Results**
178	Provide the capability for diesel driven, low pressure vessel make-up	This SAMA would provide an extra water source in sequences in which the reactor is depressurized and all other injection is unavailable (e g., FP system).	Large relative to potential averted cost-risk	The low pressure injection function is not highly important in terms of reducing risk at RNP. The largest RRW (1.006) associated with RHR (low pressure) injection failure is the common cause failure of the pump check valve to open. Pump failures have RRW values of 1 0 in both the CDF and LERF lists. The averted cost-risk associated with the installation of a diesel driven make-up system would be far less than the cost of implementation for the new system
185	Upgrade Chemical and Volume Control System to mitigate small LOCAs.	For a plant where the Chemical and Volume Control System cannot mitigate a Small LOCA, an upgrade would decrease the Small LOCA CDF contribution.	$1.7 million	The High Head Safety Injection (HHSI) system has low RRW values associated with its components The largest RRW associated with this function is the common cause failure of the suction valves from the RHR system (1.006). Common cause failure of the HHSI pumps is even lower, at 1 004. Even if it was assumed that all risk from Small LOCAs was removed through the implementation of this SAMA, the Small LOCA RRW is only 1.076 This corresponds to an averted cost-risk of about $204,000 assuming that the Small LOCA initiator affects the total maximum averted cost-risk uniformly. The averted cost-risk for this SAMA is much less than the cost of implementation
202	Passive High Pressure System	SAMA will improve prevention of core melt sequences by providing additional high pressure capability to remove decay heat through an isolation condenser type system.	$1.7 million	The cost of implementation for this modification is greater than half of the maximum averted cost-risk. Given that the RRW values for the current high head injection system are low (1.006 for CCF of the RHR path suction valves and 1.004 for CCF of the pumps), further improvements to the high pressure injection function offer limited means of reducing plant risk. In order for this SAMA to be cost effective, the RRW for the high pressure injection function would have to be closer to 2.0.

Table 4.c-1: Phase 2 Dispositions for Additional SAMAs Retained from the Phase 1 Analysis Given Use of the 95th Percentile PSA Results				
Phase 1 SAMA ID Number	**SAMA Title**	**Result of Potential Enhancement**	**Cost of Implementation**	**Phase 2 Disposition Given the Use of the 95th Percentile PSA Results**
233	Create/enhance RCS depressurization ability.	With either a new depressurization system, or with existing PORVs, head vents, and secondary side valve, RCS depressurization would allow earlier low pressure ECCS injection Even if core damage occurs, low RCS pressure would alleviate some concerns about core melt ejection	$500,000 - $4 6 million	The largest RRW value associated with secondary side depressurization, based on CDF, is 1.013 for the operator action to perform the depressurization process This action could be automated, but the benefit would be low and may introduce complexities, such as the need to inhibit the depressurization process for certain circumstances Increases in reliability due to equipment replacement would provide only a fraction of the SG PORVs' RRW of 1.013 The primary side PORVs and block valves have the highest RRWs identified for primary side depressurization, at 1.002 An entirely new system is judged to cost more than the benefit gained, and the cost of implementation is likely more than the maximum averted cost-risk of $2.89 million.

Table 4.c-2: Initial Disposition of the Original Phase 2 SAMAs and New SAMA 10 Given Use of the 95th Percentile PSA Results					
Phase 2 SAMA ID	**Averted Cost-Risk Based on All Releases**	**Averted Cost-Risk using the 95th Percentile PSA Results**	**Cost of Implementation**	**Net Value**	**Cost Beneficial Based on ER Submittal Assumptions?**
1	$0	$0	Not Required	$0	No
2	$40,392	$98,960	Not Required	N/A	No
3	$140,778	$344,906	$280,000	$64,906	Yes*
4	$0	$0	Not Required	$0	No
5	$35,893	$87,937	Not Required	N/A	No
6	$17,930	$43,929	Not Required	N/A	No
7	$32,472	$79,556	$50,000	$29,556	Yes*
8	$58,885	$144,268	$264,750	-$120,482	No
9	$0	$0	Not Required	$0	No
10	$72,083	$176,603	430,000	-$253,397	No

*These items were found not to be cost beneficial when more realistic assumptions related to the SAMA implementation were used in place of the conservative estimates included in the ER submittal.

NRC Request 5:

"Please provide the following information concerning the MACCS analyses:

a. discuss the applicability of the standard MACCS core inventory (3412 MW thermal) to RNP (2339 MW thermal), and whether the inventory was scaled to account for the lower power level,

b. please provide additional discussion to clarify what is meant by the following sentence in Section F.3.3, page F-6, "Each RNP category corresponded with a single release duration (either puff or continuous); MACCS category Te required multiple releases," and

c. the MACCS analysis assumes all releases occur at ground level and has a thermal content the same as ambient. These assumptions could be non-conservative when estimating offsite consequences. Please provide an assessment of the sensitivity of offsite consequences (doses to the population within 50 miles) to these assumptions."

CP&L Response 5.a:

The MACCS2 manual states that, "when plant-specific inventories are not available, a representative inventory may be obtained by linear scaling of the inventory of a similar reactor having a different thermal power level. The scale factor used can be chosen to be the ratio of the two reactors' thermal power levels." The standard inventory supplied with MACCS2 is for a Pressurized Water Reactor (PWR) of 3412 Megawatts thermal (MWt). The RNP power level of 2339 MWt was accounted for by setting the MACCS core inventory scaling factor to 0.686 (2339/3412).

CP&L Response 5.b:

The categories referred to are MACCS2 release categories, which are groupings of nuclides that exhibit similar physical behavior. In this case, the release categories are released either continuously, at a constant rate, or instantaneously. The phrase beginning "MACCS category Te required multiple releases" should be deleted.

CP&L Response 5.c:

The sensitivity of the assumption that the releases are at, and remain attached to, ground level was investigated by comparing the 50-mile population dose-risk that would result if the analyzed RNP scenarios were released at heights of 0, 20, 40, 60 (~top of containment) or 80 meters. It was found that the risk is relatively insensitive to this assumption. The dose-risk for release heights of 20, 40, 60 and 80 meters (relative to ground level releases) increases by 2%, 4%, 5% and 4%, respectively.

Note that 95% of the baseline risk is due to Release Categories 4C, 5C and 5. The former two (43% of risk) would likely result in releases from the SG relief valves (elevation ~13 meters above grade); the latter (52% of risk) would likely be a ground level release.

Because the effect of non-ground attached plumes is small, and the contribution to averted cost by dose-risk is small compared to replacement power and onsite cleanup costs, the effect of release elevations and thermal content would not alter the overall cost benefit conclusions.

<u>NRC Request 6:</u>

"In the Phase 2 assessment (Section F.6), the benefits associated with reducing population dose are reported in terms of percent reduction in LERF. Please provide this estimated benefit in terms of percent reduction in person-rem dose for each of the SAMAs that are quantitatively assessed."

<u>CP&L Response 6:</u>

The SAMA Phase 2 cost benefit analysis required the calculation of the change in dose-risk (person-rem/yr) for each SAMA identified; however, these changes were not reported in the ER submittal. The changes in dose-risk corresponding to each of the Phase 2 SAMAs are provided in Tables 6-1 through 6-10 for both the LERF and non-LERF release categories.

The ER submittal did not consider the non-LERF releases in the SAMA analysis, but as these contributors are comparable in magnitude to the LERF contributions, they have been included. For completeness, Table 6-11 has been included to provide the results of the Phase 2 cost benefit analysis when all releases are considered. The impact of including the non-LERF contributors was small and did not change the results of the analysis provided in the ER submittal.

In addition, Response 2.c above identified an error that was made in the initial review of the RNP Risk Reduction Worth values. The basic event "KRV%729NN" (Relief Valve CC-729 Transfers Open and Diverts Flow) was identified as a charging pump flow diversion and judged to be addressed by Phase 1 SAMA 16. This event is, in fact, a CCW system flow diversion. A new Phase 2 evaluation was performed for this SAMA.

<u>Phase 2 SAMA 10, Reduce CCW Flow Diversions</u>: The flow diversion identified for this SAMA is due to the failure of a CCW system relief valve to remain closed. The flow diverted through this valve is considered to be large enough to drain the CCW system to the point where it will not be able to perform its function.

Potential changes to reduce the risk associated with flow diversions may include the installation of a new valve with an improved design, enhanced maintenance on the valve, installation of logic to identify a failed relief valve and hardware to automatically isolate the leak, or capping of the relief valve.

Capping the relief valve is not considered to be an appropriate modification, since an important overpressure protection feature would be eliminated. Improved maintenance is a potential means of improving valve reliability, but in this case, no measurable benefit is deemed to be attainable. RNP has already implemented the Maintenance Rule, which has incorporated good maintenance practices on risk significant systems. The improvement in valve reliability associated with further enhancing valve maintenance is judged to be small and difficult to quantify with current PSA techniques. No replacement valves have been identified for CC-729 that have reliabilities that are notably better than the current valve. Some benefit may be attainable through the installation of an automatic isolation system. A differential pressure sensor could be connected to CC-729 that would identify relief valve openings and initiate isolation of the incoming CCW flow (using MOVs CC-716A and CC-716B). This would prevent the draining of the CCW system and allow time for mitigative actions to be taken. The cost benefit for this case is quantified below.

Installation of the isolation logic and related equipment is assumed to reduce the risk of this particular flow diversion by an order of magnitude. The result is a reduction in the CDF of about 5.3% (CDF_{new}=4.09E-5/yr). The changes in the dose-risk are presented in Table 6-10.

These results correspond to an averted cost-risk of $72,083. Given that the cost of implementation of this plant modification is $430,000, the net value is -$357,917. Based on the SAMA cost benefit methodology, this plant change is not cost beneficial.

Table 6-1: SAMA 1 Dose-Risk Summary

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2.33E-05	1 08E-05	2.22E-07	5.15E-06	2 62E-07	7.78E-07	3.16E-09	3.74E-08	1.81E-07	0 00E+00	3.70E-06	1.28E-06	3.94E-07
Dose-Risk (person-rem/yr)	1 01E-01	2.18E+00	2.95E-01	2 01E+00	1.64E-01	9.72E-02	6 07E-04	2.39E-02	2.79E-01	0.00E+00	1.56E+00	3.04E+00	9.38E-01
Percent Reduction from Baseline Dose-Risk	0 00%	0.00%	0 00%	0 00%	0.00%	0 00%	0 00%	0 00%	0 00%	N/A	0 00%	0 00%	0 00%

Table 6-2: SAMA 2 Dose-Risk Summary

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2 28E-05	1.07E-05	2.20E-07	5.13E-06	2.62E-07	7.78E-07	3.16E-09	3.74E-08	1.81E-07	0.00E+00	3 70E-06	1.28E-06	3.94E-07
Dose-Risk (person-rem/yr)	9.87E-02	2.16E+00	2.93E-01	2 00E+00	1 64E-01	9.72E-02	6 07E-04	2.39E-02	2.79E-01	0 00E+00	1.20E+00	3 04E+00	9 38E-01
Percent Reduction from Baseline Dose-Risk	2.28%	0.92%	0 68%	0 50%	0 00%	0 00%	0 00%	0 00%	0 00%	N/A	23 08%	0 00%	0.00%

Table 6-3: SAMA 3 Dose-Risk Summary

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2.28E-05	1.07E-05	2.20E-07	5.13E-06	2.62E-07	7.78E-07	3.16E-09	3.74E-08	1.81E-07	0.00E+00	3 70E-06	1.28E-06	3.94E-07
Dose-Risk (person-rem/yr)	9.87E-02	2.16E+00	2 93E-01	2.00E+00	1.64E-01	9.72E-02	6 07E-04	2.39E-02	2.79E-01	0 00E+00	1.56E+00	0 00E+00	7.78E-01
Percent Reduction from Baseline Dose-Risk	2.28%	0 92%	0 68%	0 50%	0 00%	0 00%	0 00%	0.00%	0 00%	N/A	0 00%	100 00%	17 06%

Table 6-4: SAMA 4 Dose-Risk Summary

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2 33E-05	1 08E-05	2.22E-07	5 15E-06	2 62E-07	7.78E-07	3.16E-09	3.74E-08	1 81E-07	0 00E+00	3 70E-06	1.28E-06	3 94E-07
Dose-Risk (person-rem/yr)	1 01E-01	2.18E+00	2.95E-01	2.01E+00	1 64E-01	9.72E-02	6 07E-04	2.39E-02	2.79E-01	0 00E+00	1.56E+00	3 04E+00	9.38E-01
Percent Reduction from Baseline Dose-Risk	0 00%	0 00%	0.00%	0 00%	0.00%	0 00%	0 00%	0 00%	0 00%	N/A	0 00%	0 00%	0 00%

Table 6-5: SAMA 5 Dose-Risk Summary

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2.19E-05	1.04E-05	2.09E-07	5.08E-06	2.59E-07	7.36E-07	3.16E-09	3.73E-08	1.80E-07	0.00E+00	3.70E-06	1.28E-06	3.94E-07
Dose-Risk (person-rem/yr)	9.48E-02	2.10E+00	2.78E-01	1.98E+00	1.62E-01	9.20E-02	6.07E-04	2.39E-02	2.77E-01	0.00E+00	1.56E+00	3.04E+00	9.38E-01
Percent Reduction from Baseline Dose-Risk	6.14%	3.67%	5.76%	1.49%	1.22%	5.35%	0.00%	0.00%	0.72%	N/A	0.00%	0.00%	0.00%

Table 6-6: SAMA 6 Dose-Risk Summary

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2.22E-05	1.06E-05	2.00E-07	5.13E-06	2.60E-07	7.78E-07	3.16E-09	3.74E-08	1.81E-07	0.00E+00	3.70E-06	1.28E-06	3.94E-07
Dose-Risk (person-rem/yr)	9.61E-02	2.14E+00	2.66E-01	2.00E+00	1.62E-01	9.72E-02	6.07E-04	2.39E-02	2.78E-01	0.00E+00	1.56E+00	3.04E+00	9.38E-01
Percent Reduction from Baseline Dose-Risk	4.85%	1.83%	9.83%	0.50%	1.22%	0.00%	0.00%	0.00%	0.36%	N/A	0.00%	0.00%	0.00%

Table 6-7: SAMA 7 Dose-Risk Summary

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2.33E-05	1.08E-05	2.22E-07	5.15E-06	2.62E-07	7.78E-07	3.16E-09	3.74E-08	1.81E-07	0.00E+00	3.70E-06	1.28E-06	6.73E-08
Dose-Risk (person-rem/yr)	1.01E-01	2.18E+00	2.95E-01	2.01E+00	1.64E-01	9.72E-02	6.07E-04	2.39E-02	2.79E-01	0.00E+00	1.56E+00	3.04E+00	1.60E-01
Percent Reduction from Baseline Dose-Risk	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	N/A	0.00%	0.00%	82.94%

Table 6-8: SAMA 8 Dose-Risk Summary

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2.33E-05	1.08E-05	2.22E-07	5.15E-06	2.62E-07	7.78E-07	3.16E-09	3.74E-08	1.80E-07	0.00E+00	2.80E-06	1.27E-06	3.62E-07
Dose-Risk (person-rem/yr)	1.01E-01	2.18E+00	2.95E-01	2.01E+00	1.64E-01	9.72E-02	6.07E-04	2.39E-02	2.78E-01	0.00E+00	1.18E+00	3.03E+00	8.62E-01
Percent Reduction from Baseline Dose-Risk	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.36%	N/A	24.36%	0.33%	8.10%

Table 6-9: SAMA 9 Dose-Risk Summary

	Non-LERF							**LERF**					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2 33E-05	1.08E-05	2 22E-07	5.15E-06	2 62E-07	7.78E-07	3.16E-09	3.74E-08	1.81E-07	0 00E+00	3.70E-06	1.28E-06	3.94E-07
Dose-Risk (person-rem/yr)	1 01E-01	2.18E+00	2.95E-01	2 01E+00	1.64E-01	9.72E-02	6 07E-04	2.39E-02	2.79E-01	0 00E+00	1.56E+00	3.04E+00	9.38E-01
Percent Reduction from Baseline Dose-Risk	0.00%	0 00%	0.00%	0 00%	0 00%	0 00%	0 00%	0 00%	0 00%	N/A	0 00%	0.00%	0 00%

Table 6-10: SAMA 10 Dose-Risk Summary

	Non-LERF							**LERF**					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency(1/yr)	2 33E-05	1 08E-05	2 22E-07	5 15E-06	2.62E-07	7 53E-07	3.16E-09	3.74E-08	1.81E-07	0 00E+00	3.70E-06	1.28E-06	6.73E-08
Dose-Risk (person-rem/yr)	1.01E-01	2.18E+00	2 95E-01	2.01E+00	1.64E-01	9.41E-02	6 07E-04	2.39E-02	2.79E-01	0.00E+00	1.56E+00	3.04E+00	1 60E-01
Percent Reduction from Baseline Dose-Risk	0.00%	0 00%	0 00%	0 00%	0 00%	3.19%	0 00%	0 00%	0 00%	N/A	0 00%	0.00%	82 94%

Table 6-11: Updated Phase 2 SAMA Results

Phase 2 SAMA ID	LERF Based Averted Cost-Risk	Averted Cost-Risk Based on All Releases	Increase in Averted Cost-Risk	Cost of Implementation	Net Value	Cost Beneficial?
1	$0	$0	0%	Not Required	$0	No
2	$39,563	$40,392	2.1%	Not Required	N/A	No
3	$140,455	$140,778	0.2%	$280,000	-$139,222	No
4	$0	$0	0%	Not Required	$0	No
5	$31,706	$35,893	13.2%	Not Required	N/A	No
6	$14,927	$17,930	20.1%	Not Required	N/A	No
7	$32,472	$32,472	0%	$50,000	-$17,528	No
8	$58,885	$58,885	0%	$264,750	-$205,865	No
9	$0	$0	0%	Not Required	$0	No
10	N/A	$72,083	N/A	$430,000	-$357,917	No

NRC Request 7:

"According to the 1997 PSA summary document (Appendix B), three of the plant improvements identified in the IPE (items 3, 9 and 10) were canceled due to cost-benefit considerations. The associated cost-benefit methodology was not described and may differ from that used in the SAMA analysis. Please provide an evaluation of the costs and benefits of these three canceled SAMAs based on the current RNP risk profile and the cost-benefit methodology described in the ER."

CP&L Response 7:

As discussed in Response 2.d above, the three plant improvements identified in the IPE that were subsequently cancelled due to cost benefit considerations are considered to have been addressed by the SAMA submittal. However, these items have been explicitly re-examined using the SAMA cost benefit methodology described within this response. The three plant improvements are:

- Item 3: Modification of the plant safety-related batteries
- Item 9: Charging pump self-cooling modification
- Item 10: Automatic re-fill of the condensate storage tank

A summary of the analysis for each of these plant improvements is provided below. These evaluations include consideration of the non-LERF contributors and implement a base case maximum averted cost-risk of $1,181,000.

Item 3, Modification of the Plant Safety-Related Batteries: The IPE identified this change to upgrade the capacity of the safety-related batteries from 1 hour to 4 hours to allow more time for offsite power recovery. SAMA 92 specifically addresses increasing the battery capacity for RNP. This SAMA was screened in Phase 1, since the estimated cost of implementation was greater than the RNP maximum averted cost-risk. Given the addition of the non-LERF contributions to the cost benefit model and consideration of a lower cost alternative to improve DC capability, a re-examination of this SAMA is warranted.

The original plant change suggested replacement or modification of the safety-related batteries to increase the DC capacity in an SBO. As indicated in Response 8, there are other means of increasing the plant's DC capacity for lower costs. Specifically, a portable DC charger that can be connected to the plant's DC bus in an SBO would provide DC power for a potentially unlimited duration. Compared with the 3 hour increase in DC availability afforded by the battery upgrade, the portable charger is more desirable.

Given that a safety-related emergency diesel generator has a combined start/run failure probability of 3.4E-2, it was considered appropriate to apply a similar reliability to the portable generator. A lumped system failure of 1E-2 was applied to the charger to account for hardware and operator failures. The action OPER-BC, aligning charger to the bus after an undervoltage trip, was not applied to the portable charger as a common failure for all chargers. This was excluded due to the fact that this action applies to the normal and alternate chargers when there is still AC power available to the chargers. The dominant sequences for the portable charger are SBO sequences, which are judged to impose different operator cues.

Operations staff and Human Resources personnel have been consulted on the operator task loading and resources required to support the addition of a portable DC charger. Since operators are already assigned to tasks in the scenarios in which the DC charger would be required, a new operator position would need to be created to support DC charger manipulation. Based on annual salary requirements and benefits, the cost of implementing this SAMA is $350,000 per year. This estimate assumes a cost of $70,000 per year and the need for an additional operator on each of the five crews. No costs related to crew training, procedure modifications, hardware modifications, maintenance program modifications, or the cost of the charger itself have been included. Since this change would be required to be in effect for each of the 20 years of the renewed license, the total salary-based cost would be $7,000,000 in 2002 dollars.

Incorporation of the portable DC charger into the model resulted in a reduction in CDF of about 4.2% (CDF_{new}=4.14E-5/yr). The changes in the dose-risk and offsite economic cost-risk (OECR) are presented in Table 7-1.

Table 7-1: Level 3 Results for Modification of the Plant Safety-Related Batteries

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency (1/yr)	2 21E-05	1 06E-05	1.96E-07	4 56E-06	1 95E-07	7 68E-07	3.16E-09	1 84E-08	1.74E-07	0 00E+00	3.70E-06	1 28E-06	3.94E-07
Dose-Risk (person-rem/yr)	9 57E-02	2.14E+00	2 61E-01	1.78E+00	1 22E-01	9 60E-02	6 07E-04	1.18E-02	2 68E-01	0 00E+00	1.56E+00	3.04E+00	9 38E-01
Percent Reduction from Baseline Dose-Risk	5.25%	1.78%	11 63%	11 52%	25 80%	1 23%	0 05%	50 73%	3.96%	N/A	0 17%	0 00%	0 00%
OECR	3 45E+00	8.85E+02	6 39E+02	1 73E+03	3 28E+02	2.80E+01	3 19E-01	2 08E+01	6 93E+02	0.00E+00	3 08E+03	4 35E+03	1 34E+03
Percent Reduction from Baseline OECR	5 15%	1.85%	11.71%	11 46%	25 57%	1.29%	0 00%	50 78%	3 97%	N/A	0 00%	0 00%	0 00%

These results correspond to an averted cost-risk of $46,946. Given that the cost of implementation of this plant modification is $7,000,000, the net value is -$6,953,054. Based on the SAMA cost benefit methodology, this plant change is not cost beneficial.

<u>Item 9, Charging Pump Self-Cooling Modification:</u> This item would remove the charging pump cooling dependence on CCW by installing new, self-cooled pumps or modifying the current pumps to be self-cooled. While more cost beneficial means of providing cooling to the charging pumps on loss of CCW have been implemented (i.e., procedures developed to align alternate cooling), the self-cooling modification would eliminate the reliance on human action to ensure cooling.

The RNP PSA model was changed to remove the primary CCW dependence and incorporate an independent radiator system (self-cooling). The radiator system was represented as a lumped event with a failure probability of 1E-3. Each pump train is equipped with its own, independent system. Common cause failure contributions were not explicitly considered in the application of the events, which will show increased benefit for the self-cooling modification. The CCW cooling function was retained in the model as an alternate cooling method in conjunction with the SW and fire water connections.

Implementation of the charging pump self-cooling modification is estimated to yield a reduction in CDF of about 35% (CDF_{new}=2.82E-5/yr). The changes in the dose-risk and offsite economic cost-risk (OECR) are presented in Table 7-2.

Table 7-2: Level 3 Results for Charging Pump Self-Cooling Modification

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency (1/yr)	1 66E-05	4.08E-06	2 01E-07	4 18E-06	2 60E-07	2 73E-07	3.16E-09	9 19E-09	1 22E-07	0 00E+00	3 70E-06	1 28E-06	3.94E-07
Dose-Risk (person-rem/yr)	7.19E-02	8.24E-01	2 67E-01	1 63E+00	1 62E-01	3 41E-02	6 07E-04	5 88E-03	1 88E-01	0 00E+00	1 56E+00	3 04E+00	9.38E-01
Percent Reduction from Baseline Dose-Risk	28 83%	62.19%	9 38%	18 90%	1 07%	64 89%	0 05%	75 39%	32 66%	N/A	0 00%	0 00%	0 03%
OECR	2.59	340 68	655 26	1,588 40	436 80	9 94	0 32	10 38	485.56	0 00	3,081.27	4,345.20	1,339 60
Percent Reduction from Baseline OECR	28 76%	62.22%	9 46%	18 83%	0.76%	64 91%	0 00%	75.41%	32 67%	N/A	0 00%	0 00%	0 03%

These results correspond to an averted cost-risk of $335,550. Given that the cost of implementation of this plant modification is $682,000, the net value is -$346,450. Based on the SAMA cost benefit methodology, this plant change is not cost beneficial.

Item 10: Automatic Re-Fill of the Condensate Storage Tank: This modification addresses the dependence on operator action for scenarios requiring long-term availability of the CST. Capability would be added to automatically initiate the re-fill process in order to maintain a useable CST level. In this case, it is proposed that a vacuum breaker be installed between the Service Water System and the CST, such that inventory would automatically be restored on low level. It is assumed that the existing pumps can meet the flow requirements.

This plant enhancement has been modeled through the use of existing model structures to approximate the impact of this modification. A lumped event representing the automatic re-fill function has been combined with the operator actions to perform the manual re-alignment of the AFW pump suction to the alternate sources. The failure probability of the event is assumed to be 1E-4. The operator actions previously used as the primary means of make-up initiation have been retained as back-up actions for the automatic function. Currently, the makeup sources credited include the Service Water and Deep Well pumps, which address the primary sequences where long term secondary side heat removal is important.

Implementation of the automatic CST re-fill modification is estimated to yield a reduction in CDF of about 6.5% (CDF_{new}=4.04E-5/yr). The changes in the dose-risk and offsite economic cost-risk (OECR) are presented in Table 7-3.

Table 7-3: Level 3 Results for Automatic Re-Fill of the Condensate Storage Tank

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency (1/yr)	2 17E-05	1 02E-05	1.56E-07	5 08E-06	2 54E-07	7.51E-07	3.16E-09	2 81E-08	1.77E-07	0 00E+00	2 47E-06	1.28E-06	3 94E-07
Dose-Risk (person-rem/yr)	9 40E-02	2 06E+00	2 07E-01	1.98E+00	1 58E-01	9 39E-02	6 07E-04	1.80E-02	2.73E-01	0 00E+00	1 04E+00	3 04E+00	9.38E-01
Percent Reduction from Baseline Dose-Risk	6 97%	5 49%	29.67%	1 43%	3 36%	3 42%	0.05%	24 75%	2 30%	N/A	33.34%	-0 05%	0 03%
OECR	3 39	851 70	508 56	1,930 40	426.72	27.34	0 32	31.75	704.46	0 00	2,057 51	4,345 20	1,339 60
Percent Reduction from Baseline OECR	6 87%	5 56%	29 73%	1.36%	3.05%	3 47%	0 00%	24 83%	2.32%	N/A	33 23%	0 00%	0 03%

These results correspond to an averted cost-risk of $75,305. Given that the cost of implementation of this plant modification is $484,000, the net value is -$408,695. Based on the SAMA cost benefit methodology, this plant change is not cost beneficial.

NRC Request 8:

"for certain SAMAs considered in the ER, there may be lower cost alternatives that could achieve much of the risk reduction. In this regard, please provide the following:

a. for the subset of plant-specific SAMAs identified in RAI 2d and for the Phase 2 SAMAs, discuss whether any lower-cost alternatives to those considered in the ER would be viable and potentially cost-beneficial,

b. SAMAs 92 and 93 address added DC capability with costs estimated as being greater than $1.8M, thus, eliminating them from further consideration. Please provide the averted-risk benefit from these SAMAs, and address whether less costly alternatives to the SAMAs suggested might make these alternatives viable. Specifically consider and provide estimated costs and benefits for diesel-driven battery chargers, and cross-connects to the existing non-safety station batteries as two potential alternatives,

c. a plant has recently installed a direct-drive diesel to power an AFW pump for under $200K. Please provide the averted-risk benefit of supplemental AFW capability at Robinson, and an assessment of whether such a SAMA could be a cost-beneficial alternative to a motor-driven pump (Phase 1 SAMA 176), and

d. please provide an assessment of the costs and benefits of an automatic safety injection pump trip on low refueling water storage tank level as an alternative to fully automating the switch-over from injection to recirculation (Phase 2 SAMA 8)."

CP&L Response 8.a:

"... discuss whether any lower-cost alternatives to those considered in the ER would be viable and potentially cost-beneficial,"

No viable lower-cost alternatives have been identified.

CP&L Response 8.b:

"SAMAs 92 and 93 address added DC capability with costs estimated as being greater than $1.8M, thus, eliminating them from further consideration. Please provide the averted-risk benefit from these SAMAs, and address whether less costly alternatives to the SAMAs suggested might make these alternatives viable. Specifically consider and provide estimated costs and benefits for diesel-driven battery chargers, and cross-connects to the existing non-safety station batteries as two potential alternatives,"

Please refer to Response 7, above.

CP&L Response 8.c:

"a plant has recently installed a direct-drive diesel to power an AFW pump for under $200K. Please provide the averted-risk benefit of supplemental AFW capability at Robinson, and an assessment of whether such a SAMA could be a cost-beneficial alternative to a motor-driven pump (Phase 1 SAMA 176),"

Incorporation of the direct-drive AFW pump into the model resulted in a reduction in CDF of about 9.7% (CDF_{new}=3.90E-5/yr). The changes in the dose-risk and offsite economic cost-risk (OECR) are presented in Table 8-1.

Table 8-1: Level 3 Results for Response 8.c (Incorporation of Direct-Drive AFW Pump)

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency (1/yr)	2.24E-05	1 05E-05	1 91E-07	3 03E-06	2 67E-08	0 00E+00	0 00E+00	9.50E-11	1.12E-07	0 00E+00	3.18E-06	1 27E-06	3 94E-07
Dose-Risk (person-rem/yr)	9.70E-02	2.12E+00	2 54E-01	1.18E+00	1.67E-02	0 00E+00	0 00E+00	6 08E-05	1.72E-01	0 00E+00	1.34E+00	3 02E+00	9 38E-01
Percent Reduction from Baseline Dose-Risk	3.97%	2.71%	13 89%	41 21%	89.84%	100 00%	100 00%	99.75%	38.18%	N/A	14 18%	0 57%	0 03%
OECR	3 49	876.75	622 66	1,151 40	44 86	0.00	0 00	0 11	445.76	0 00	2,648 94	4,318 00	1,339 60
Percent Reduction from Baseline OECR	3.86%	2 78%	13 96%	41.17%	89 81%	100 00%	100 00%	99.75%	38.19%	N/A	14 03%	0 63%	0 03%

These results correspond to an averted cost-risk of $134,510. Given that the cost of implementation of this plant modification has been estimated to be $200,000, the net value is -$64,490. Based on the SAMA cost benefit methodology, this plant change is not cost beneficial.

<u>CP&L Response 8.d:</u>

"please provide an assessment of the costs and benefits of an automatic safety injection pump trip on low refueling water storage tank level as an alternative to fully automating the switch-over from injection to recirculation (Phase 2 SAMA 8)."

Due to more limited function, implementation of the automatic pump trip on low RWST level will have an averted cost-risk that is less than full automation of the switch-over to recirculation. Fully automated switch-over yields an averted cost-risk of about $59,000, as documented in Response 6. This corresponds to a reduction in CDF of about 4.9% (CDF_{new}=4.11E-5/yr). The Level 3 results are presented in Table 8-2. Similar to other plants, CP&L assumes a minimum cost to implement a hardware change of $70,000 and a minimum cost for a procedure change, including training, of $30,000. Since the cost of implementation of the pump trip logic is greater than the averted cost-risk for full automation of the switch-over action, the automatic pump trip is not a cost beneficial enhancement.

Table 8-2: Level 3 Results for Response 8.d (Low RWST Level Pump Trip)

	Non-LERF							LERF					
Release Category	IC-1	RC-1	RC-1A	RC-1B	RC-1BA	RC-3	RC-3B	RC-2	RC-2B	RC-4	RC-4C	RC-5	RC-5C
Frequency (1/yr)	2.33E-05	1 08E-05	2 22E-07	5.15E-06	2 62E-07	7 78E-07	3.16E-09	3.74E-08	1.80E-07	0 00E+00	2 80E-06	1.27E-06	3.62E-07
Dose-Risk (person-rem/yr)	1 01E-01	2.18E+00	2 95E-01	2 01E+00	1 64E-01	9.72E-02	6 07E-04	2 39E-02	2.78E-01	0.00E+00	1.18E+00	3 03E+00	8 62E-01
Percent Reduction from Baseline Dose-Risk	0 00%	0 00%	0 00%	0 00%	0 00%	0 00%	0 00%	0 00%	0 36%	N/A	24.36%	0 33%	8 10%
OECR	3 63	901.80	723 72	1,957 00	440 16	28.32	0 32	42 24	717 59	0 00	2,329 07	4,328 20	1,231.48
Percent Reduction from Baseline OECR	0 00%	0 00%	0 00%	0 00%	0 00%	0 00%	0 00%	0 00%	0 50%	N/A	24 41%	0 39%	8 09%

NRC Request 9:

"The RNP PRA does not utilize the Rhodes reactor coolant pump (RCP) seal LOCA model endorsed by the NRC. The use of this model could impact the risk from RCP seal LOCA events and the estimated benefits of associated SAMAs. Please discuss the RCP seal LOCA model used in the PSA and why this is judged to provide an appropriate representation of RCP seal LOCA events. Provide an assessment of the potential impact that use of the Rhodes model could have on the cost-benefit results for those SAMAs associated with RCP seal LOCAs. Also, provide an estimate of when RCP seals constructed of improved materials will be installed on pump "A" (see Phase 1 SAMA 14)."

CP&L Response 9:

CP&L will provide a schedule by January 15, 2003, for responding to NRC Request 9.

NRC Request 10:

"For SAMAs 59 and 60 -- SAMAs that have already been implemented at Robinson -- reference is made in Table F-8 to the suppression pool in discussions of the enhancements. Please explain the relevance of suppression pools to the SAMAs under consideration. Also, clarify the reference to suppression pools in the discussion of SAMA 116."

CP&L Response 10:

Part of the philosophy in generating the SAMA list was to maintain the text description of the SAMA as it was presented in the source document. This text is included in the columns for the "SAMA Title" or "Result of Potential Enhancement" in Table F-8 of the ER. This was done to preserve the means of identifying the intent of the SAMA as it was originally written. While many of the SAMAs were created for plants with different types of equipment or structures than those at RNP, it was possible to functionally link the SAMAs to RNP.

NRC Request 10 specifically questions the references to the suppression pool in SAMAs 59, 60, and 116 of Table F-8, and how the disposition "Already Implemented at Robinson" could be applied to a plant without a suppression pool. The rationale behind the disposition of these SAMAs is discussed below:

SAMA 59, Re-Fill CST: The result of this enhancement is described as providing a cool suction source for AFW pumps in SBOs or LOCAs. For BWR SBO cases, suppression pool cooling is not available due to power dependencies, and maintaining suction on the CST will allow continued pump operation. This enhancement is beneficial for LOCAs in scenarios where suppression pool cooling has been disabled due to strainer clogging or other suppression pool cooling failures. Given that the suppression pool is not available, use of the CST allows continued injection. While this SAMA was developed for a BWR, it was reviewed to determine if there was a functional equivalent at RNP.

It was determined that for SBO cases, an equivalent type of improvement would be to provide an alternate means of supplying water to the AFW pump when the normal CST inventory was depleted. A connection to Service Water was identified, but it is AC dependent and not useful in an SBO. RNP has implemented abnormal operating procedures that allow use of the diesel fire pump to provide makeup to the CST under SBO conditions. This was considered to address the SBO portion of SAMA 59.

For LOCA cases, the sump is the injection system suction source when the RWST has been depleted. If the RHR cooling function fails, the sump will heat-up and the suction source may be lost due to pump cavitation. Re-fill of the RWST would address this issue for RNP in LOCA conditions. SAMA 182 addresses this specific topic.

SAMA 60, Maintain ECCS Suction on CST: This SAMA is also related to loss of the injection suction source due to heat-up. The disposition for this SAMA addresses the use of the CST for make-up to the steam generators.

For this SAMA, a more appropriate functional link would have been maintaining the RWST as the RHR suction source for as long as possible to prevent sump inventory heat-up in the event that cooling is not available. This would provide the operators with more time to restore the RHR cooling function.

In this case, End Path Procedure (EPP) –9, "Transfer to Cold Leg Recirculation," provides explicit instructions on the level at which the switch-over to re-circulation mode is to be made. These instructions are already considered to be optimized to allow the proper amount of time for the action. There is a balance that must be maintained between providing additional time for system recovery and the need to avoid damaging equipment that may be required later in the accident.

SAMA 116, Steam Driven Turbine Generator: The use of "suppression pool" in the "Result of Potential Enhancement" column is due to the retention of the original SAMA text, as mentioned above. It does not reflect the manner in which the enhancement would be implemented at RNP.

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/02-0180
Seven Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

APPENDIX A FROM PROBABILISTIC SAFETY ASSESSMENT SUMMARY DOCUMENT – 1997

List of Acronyms

HBRSEP2	H. B. Robinson Steam Electric Plant, Unit No. 2
IPE	Individual Plant Examination
PSA	Probabilistic Safety Assessment
LOCA	Loss of Coolant Accident
EPP	End Path Procedure
CST	Condensate Storage Tank
MAAP	Modular Accident Analysis Program
RCP	Reactor Coolant Pumps
LOOP	Loss of Offsite Power
ECCS	Emergency Core Cooling System
MOV	Motor Operated Valve
S/G	Steam Generator
EDG	Emergency Diesel Generator
SBO	Station Blackout
RCS	Reactor Coolant System
PORV	Power Operated Relief Valve
SRV	Safety Relief Valve
AFW	Auxiliary Feedwater
SDP	Steam Driven Pump
MDP	Motor Driven Pump
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SI	Safety Injection
DC	Direct Current
AOP	Abnormal Operating Procedure
CCW	Component Cooling Water
CDF	Core Damage Frequency
MFW	Main Feedwater
HRA	Human Reliability Analysis
EPRI	Electric Power Research Institute

Note: The purpose of the Probabilistic Safety Assessment (PSA) Summary Document is to provide an overview of the technology of PSA, and to summarize the details, results and potential applications of the H. B. Robinson Steam Electric Plant, Unit No. 2, PSA model. This document provides a concise summary of the important features and conclusions of the updated PSA analysis, and gives those who may wish to use the PSA a general understanding of the process and how it may be applied.

APPENDIX A - Model Updates Since the HBRSEP2 IPE Submittal

Since the IPE Submittal, the HBRSEP2 PSA model has undergone an update that incorporates new plant-specific data, procedural changes and plant modifications. This update also reflects new technology in PSA methodology developed since the IPE submittal. The following changes to the PSA model had the greatest reduction on the total core damage frequency:

LOCA probability. The LOCA frequencies used in the IPE were based on frequencies taken from seven other PSAs. Limited industry failure data had led to conservative LOCA initiator frequencies. Methodology, developed by EPRI (EPRI TR-102266, "Pipe Failure Study Update") to compute LOCA initiating event frequencies based on plant specific parameters, was incorporated into the HBRSEP2 PSA.

Latent human interaction. The assumptions used for screening criteria for identification of latent human interactions were revised. Based on these new screening criteria, as well as procedural enhancements, several latent human interactions were removed from the PSA model.

Flooding Procedures Update. The new and revised flooding procedures were written to assist the operator in identifying sources of flooding and potential isolation measures. In addition, these procedures were written to limit the accumulation of water thereby limiting the potential for equipment damage. These procedure updates were used in reassessing the PSA flooding model to specifically credit the instructions to open outside doors to egress water.

Addition of Dedicated Shutdown Diesel Generator (DSDG) to EPPs. A new procedure was written, EPP-22, "Energizing Plant Equipment Using Dedicated Shutdown Diesel Generator," to direct the operating crew to align equipment as needed to the Dedicated Shutdown (DS) Bus and not to limit the use of the DS Bus to Appendix R or Station Blackout conditions. For the IPE, a recovery event was manually applied to station blackout cutsets. The PSA model was updated to include the DSDG system components and operator actions to perform this new procedure.

CST refill analysis for 80% nominal average. Historical data were reviewed to determine the actual level that was maintained in the CST. The average available inventory was greater than the inventory assumed in the IPE. An analysis using the MAAP code predicted the time for core uncovery based on CST depletion with no makeup. The MAAP analysis indicated that, with the RCPs not running, CST inventory was sufficient to assure core integrity past 24 hours. This was a sufficient time for recovery, and therefore, no CST depletion and refill failures were modeled with Loss of Offsite Power sequences. This update reduced LOOP sequence risk.

ECCS MOVs Update. At the time of the IPE, ECCS MOV active failures were modeled as non-time dependent demand failures. Because many of these MOVs were only stroke tested once per refueling cycle, their demand failure probability value was increased based on the time dependent standby failure rate and the exposure time.

Frequencies of transient initiating events update. The transient initiating event frequencies were updated to reflect the most recent operating experience through 1995. The IPE evaluated many years of operational data to develop plant specific transient initiating event data. The transient initiating event frequencies were updated because HBRSEP2 had experienced a significant reduction in plant trips following the replacement of the S/Gs in 1984. The update lowered the transient initiating event frequency as a whole.

Frequency of Loss of Offsite Power. The LOOP initiating event analysis was updated as part of the transient event update, because HBRSEP2 had experienced a LOOP in 1992, and because the EPRI TR-106306, "Losses of Off-Site Power at U.S. Nuclear Plants - Through 1995," was available. This revision incorporated a change in the methodology used to determine the LOOP frequency compared to the methodology used in the IPE. The new methodology counted specific LOOPs per unit rather than per site. (This change was first incorporated in NSAC-166, "Losses of Off-Site Power at U.S. Nuclear Power Plants - Through 1990." According to this document, for industry LOOP events from 1980 to 1990, the LOOP per unit year frequency was 0.035 and the per site year frequency was 0.047. The most current EPRI document, EPRI TR-106306 indicated that the industry value for LOOP per unit year was 0.036 for years 1980 through 1995. This value was essentially equivalent to the plant specific Bayesian updated LOOP frequency. The resultant plant specific LOOP initiating event frequency represented an overall reduction over the IPE value.

Update of Plant Specific Data. Plant specific data for major pumps and diesels were updated through a review of the control operator logbooks and work tickets. Data used in the IPE covered operational history from 1985 through 1990. The first update brought initiating event data and EDGs up to date through 1992. A more current update brought the plant specific data up to date through 1995.

Diesel Generator Data update. The plant specific diesel start and run data were updated in order to remove conservatisms from the data used in the IPE Submittal. For the IPE, any diesel trip was counted as a failure. A majority of the trips experienced during testing were due to faulted trip signals that were defeated during normal standby operation. Under normal operational conditions, the diesel would not have tripped and would have continued to run without any problems. The data update screened out the non-applicable trips and used only actual diesel run and start failures. This update lowered SBO risk.

RCS Pressure Challenge Update. Unavailability of the RCS PORVs to mitigate a pressure challenge may lead to a RCS SRV challenge. Challenges to the RCS SRVs increases the likelihood of an SRV LOCA. Two updates were performed to remove conservatisms associated with transient induced SRV LOCAs. The first update was a review of the operational history of running with closed block valves. Improvements in the method of blocking open the PORVs during refueling had virtually eliminated the occurrence of PORV leakage and subsequent block valve closure. Thus, the conditional probability of PORV block valve closure in the IPE model was updated accordingly.

The second model update involved removal of battery depletion events from the RCS PORVs support system fault trees following a LOOP event. The IPE model allowed battery depletion to fail power to the PORVs irrespective of timing considerations. Battery depletion was modeled to occur 1 hour after a loss of the battery chargers and not fail the PORVs at the time of the transient and subsequent pressure challenge. The model was updated accordingly to give the net effect of a reduction in LOOP contribution.

AFW Common Cause Failure Update. The IPE model included a common cause failure of all three AFW pumps based on other reference PSAs. A determination was made that since there was no similarity in design, manufacturer, or location of the AFW SDP, and that this common cause event was invalid. The PSA model was updated to only include a common cause failure of the AFW MDPs.

Update of Cold Leg Recirculation Model. At the time of the IPE, EPP-9, "Transfer to Cold Leg Recirculation," was written such that, for any size LOCA, the high pressure SI pumps were always aligned to take suction from the RHR pumps during recirculation mode. The procedure was revised such that, for any size LOCA, low head recirculation was initiated first with one SI pump continuing to take suction from the RWST flow until low pressure flow to the RCS could be verified. For inadequate low head flow, the SI pump was stopped to establish the recirculation path and then restarted to initiate high head recirculation. Thermal hydraulic analyses indicated that high head recirculation would only be required for small break LOCAs with size ranging from 3/8" to 5". Breaks sizes larger than 5 inches, medium and large LOCAs, were determined to be mitigated by low pressure recirculation. The PSA was updated to remove the requirement for high head pumps during recirculation for medium and larger break LOCAs.

Update of Loss of DC Bus Events. A review of new and existing procedures was made to evaluate the most current procedural response to a loss of DC Bus. Two recoveries were found and included in the model. The first was the ability to recover a loss of AFW flow by manually closing the Emergency Bus E-2 output breaker for AFW pump B following a loss of DC Bus B. The second recovery was the ability to locally open the Main Feedwater Regulator by-pass valves to establish feed flow from the available Feedwater/Condensate train after a loss of either DC Bus A or B. Both recoveries were included in procedure FRP-H.1, "Response to Loss of Secondary Heat Sink."

Addition of Charging Pump C. The IPE only credited two charging pumps available, pumps A and B. To be more realistic, the PSA model was updated to include all three charging pumps.

Incorporation of AFW steam-driven pump modification. The steam-driven AFW pump was permanently aligned for self-cooling mode as part of a plant modification. Prior to this change, procedures had the operator place the pump in self-cooling mode upon a loss of service water cooling. The IPE included failure of the operator to perform this action as a pump failure mechanism. The model was updated to remove this action.

Alternate cooling of SI pumps with fire water. AOP-22, "Loss of Service Water," was revised to instruct the operator to use fire water as alternate cooling for the thrust bearing coolers on a loss of service water. The PSA model was updated to include the firewater components and operator actions. Because service water was required to cool CCW during

ECCS recirculation mode, the model update was not credited to reduce CDF but only to delay core damage from the injection mode to the recirculation mode for Level II consideration

Alternate cooling to AFW MDPs with fire water. AOP-22, "Loss of Service Water," was revised to instruct the operator to use fire water as an alternate means for cooling the AFW MDP oil coolers on a loss of service water. In the IPE, a recovery action was manually added to the cutsets based on engineering judgment. The PSA model was updated to include the firewater components and the operator action for dependency assessment.

SI reset switch for Main Feedwater. The modification added a key lock switch on the RTGB to override the SI signal relays for the MFW system. Previously, the operator had to hold down the SI reset button until all the safeguards relays were defeated by removing power. This time consuming action was considered not feasible during the HRA analysis for the IPE and was not credited. With completion of this modification, the PSA was updated to credit use of MFW under scenarios that actuate an SI signal.

Addition of high head SI pumps recirculation line strainers. A modification was performed to add strainers to each SI pump's recirculation line to the RWST in order to prevent foreign materials from plugging the recirculation line flow element. The PSA model was updated to include a potential plugging event of these strainers.

Alternate compressed gas supply for S/G PORVs. A modification was installed to cross connect the S/G PORV instrument air header to the steam dump nitrogen accumulator. The PSA model was updated by adding steam dump nitrogen components and operator action to recover S/G PORVs on a loss of instrument air.

Containment Isolation Failure Revision. Two changes were made to the containment isolation fault tree. The first change, also discussed above, was revision of the latent human errors affecting the containment spray system. These latent human errors were a containment bypass mechanism following containment spray pump failure or SBO. The second change was a removal of the conservatism that a failure of either containment manway door would fail containment isolation. The success criterion was changed to require failure of both doors for containment isolation failure.

Core Debris Cooling. The IPE used information contained in Generic Letter 88-20. Of interest was the conclusion that debris cooling was assured if the debris thickness was less than 25cm and water covered the debris. More recent experimental information tended to increase the uncertainty of this conclusion. A more conservative debris cooling model was developed and incorporated into the level 2 model.

Removal of Induced Steam Generator Tube Rupture. The level 2 model was updated to remove the contribution to containment bypass resulting from using the reactor coolant pumps as a last attempt to cool the core. To this change was attributed a significant reduction in the probability of an induced SG tube rupture during a severe accident. This change was initiated because HBRSEP2 Functional Restoration Procedure (FRP)-C.1, "Response to Inadequate Core Cooling," has been revised to remove steps to restart the reactor coolant pumps. This revision stems from the implementation of the Severe Accident Management Guidelines at HBRSEP2.

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/02-0180
28 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

CALCULATION RNP-F/PSA-0001

(without Attachments)

SYSTEM FILE NO. 9400

CALC. TYPE PSA

CATEGORY B

CAROLINA POWER & LIGHT COMPANY

RNP-F/PSA-0001

FOR

Updated Individual Plant Examination Probabilistic Safety Assessment Model

FOR

H.B. ROBINSON - UNIT NO. 2

SAFETY RELATED: ☐

AUGMENTED QUALITY ☐

NON SAFETY ☒

NON SAFETY RELATED - BEYOND DESIGN BASIS ☐

APPROVAL

Rev No.	Prepared By Date	Cross Discipline Review By Discipline/Date	Review By Date	Supervisor Date
0	Bruce A. Morgen 3/19/99 Steven. A. Laur 3/19/99	N/A	S. L. Mabe 3/19/99 Steven. A. Laur 3/19/99	Steven. A. Laur 3/19/99
Reason for Change: Document the RNP Updated IPE PSA model for future reference.				
1	Bruce A. Morgen 7/21/00 Steven A. Laur 7/24/00	N/A	Bradley W. Dolan 7/21/00 Steven A. Laur 7/24/00	Steven. A. Laur 7/24/00
Reason for Change: Document changes through RFO19.				
Reason for Change:				
Reason for Change:				

LIST OF EFFECTIVE PAGES

PAGE	REV	PAGE	REV	PAGE	REV
Cover	1				
2-28	1				

ATTACHMENTS	
A-AA	1

REV	DATE	REVISION HISTORY
0	03/19/99	Revision 0 provides the baseline documentation for the RNP Updated IPE PSA model based on plant configuration as of Refueling Outage 17 and plant specific data as of December 1995.
1	7/24/00	Revision 1 provides the baseline documentation for the RNP Updated IPE PSA model based on plant configuration as of Refueling Outage 19.

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this calculation is to document the Probabilistic Safety Assessment (PSA) for the Updated Individual Plant Examination (IPE) Model for the Robinson Nuclear Plant. The Model is based on plant configuration as of Refueling Outage 19 and is termed MOR99. Reference to the Model will be by referencing the calculation number and revision.

2.0 LIST OF REFERENCES

2.1 Carolina Power & Light Company, H. B. Robinson Steam Electric Plant Unit No. 2, Individual Plant Examination Submittal, August 1992.
2.2 Calculation RNP-F/PSA-0008, Recovery Rule File for Model of Record, Revision 0.
2.3 Calculation RNP-F/PSA-0009, Assessment of Internally Initiated Flooding Events, Revision 0.
2.4 Calculation RNP-F/PSA-0010, Update of RHR Suction Line ISLOCA Sequence Analysis, Revision 0.

3.0 ENGINEERING ANALYSIS SOFTWARE

3.1 Computer Codes Used

The following computer software was used for this calculation.

SOFTWARE NAME	SOFTWARE VERSION NO.	SOFTWARE OWNER	EFFECTIVE DATE	OUTPUT FILENAME
CAFTA-W	3.2b	Steven. A. Laur	12-May-97	N/A
NURELMCS	3.1a	Steven. A. Laur	15-Dec-95	N/A
PRAQUANT	3.3	Steven. A. Laur	12-Mar-98	N/A
QRECOVER	1.4e	Steven. A. Laur	04-Jan-00	N/A

The software listed above was qualified as Software Quality Level C per CSP-NGGC-2503. The above software is also listed on the Calculation Matrix Reference Table, Attachment A.

A benchmark calculation of baseline CDF was performed with the copy of the previous model of record (MOR97) to verify the proper operation of the above software. A baseline CDF of 4.927E-5 was obtained, which agreed with the model of record. Cutsets generated in the calculation were compared to with the baseline model of record cutsets and were found to be in agreement. Attachment Y provides the benchmarking results (top 50 cutsets).

3.2 Computers Used

The above software is installed on the NGG standard desktop and may be accessed from any network PC equipped with a CAFTA hardware key.

4.0 BODY OF CALCULATION

4.1 Design Inputs

The Updated IPE PSA Model does not provide plant design basis information nor is the Model used to modify design outputs. Therefore, no design inputs were used for this Model.

The following inputs were used to create MOR99:
1. The pre-existing model, MOR97 (Calculation RNP-F/PSA-0001, Rev0).
2. Assessment of Internally Initiated Flooding Events, Revision 0, March 1998, Report RSC 97-26 (Calculation RNP-F/PSA-0009, Rev0).
3. Update of RHR Suction Line ISLOCA Sequence Analysis, Revision 1, Report RSC 99-06 (Calculation RNP-F/PSA-0010, Rev0).
4. The updated text-based recovery rule file, L1_RUL99.TXT (Calculation RNP-F/PSA-0008, Rev0).
5. Model changes, as described in this calculation.

4.2 Assumptions

The IPE PSA Model and assumptions are described in the documentation prepared for the IPE submittal (Reference 2.1).

4.3 Calculations

4.3.1 Each of the twenty-four files (24) files comprising MOR99 (listed in Attachment B), was printed in its entirety. In some cases the file was printed in sections to maintain legibility; however, the entire contents were printed. Note that some files are unchanged from MOR97; in these cases, the files were copied into the new directory. Some files are no longer used, in those cases the files were not added to the directory.

The files are maintained in the NFM&SA controlled library. Upon approval of this calculation, these files will be placed in location:

Nt000229(P):\Site\HNPApps\CONTROL\DOCUMENT\PSA\#MODELS.MOR\RNP\MOR99

4.3.2 Event Tree Changes

The event trees for the PSA Updated Individual Plant Examination (IPE) Model for the Robinson Nuclear Plant, termed MOR97, were created prior to the determination that PSA work would be prepared in accordance with engineering procedures and applicable portions of the CP&L Appendix B Quality Assurance program and would be maintained under configuration control. Accordingly, the event trees were "grandfathered" (meaning the event trees and associated pre-existing documentation has been accepted for use "as is" without further validation or review).

There were no event tree changes for MOR99. Accordingly, the pre-existing event trees remain applicable for MOR99. The following events comprise the MOR99:

- Large LOCA (EA.ETA)
- Medium LOCA (EM.ETA)
- Small LOCA (ES1.ETA and ES2.ETA)
- Anticipated Transient Without Scram (EATWS.ETA)
- Steam Generator Tube Rupture (ER.ETA)
- Transients (ET.ETA)
- Transient-Induced LOCA (ETQ.ETA)
- Bridge (EBRIDGE.ETA)
- Loss of Offsite Power (ELOSP.ETA and ETQLOSP.ETA)
- Flooding (ETFLOOD.ETA and ETQFLOOD.ETA)

Printouts from the above files are included as Attachment Z.

4.3.3 Model Changes

<u>Plant configuration changes and model improvements</u>

The following changes were applied to RNPSYSTM.CAF.

MOR97 COMMENT AND INVESTIGATION	**MOR99\RNPSYSTM.CAF ACTION TAKEN**
Revise logic to no longer use modules.	All gates with the property of "module" were identified and the property de-selected.
Change "3TBMFWBSI" to	Typo corrected.

MOR97 COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
"#TBMFWBSI.	
Gate W222 should not appear below W220." Only north header supplies circ water seals.	Removed gate W222.
The current model does not include the partial loss of feedwater as a means for failing main feedwater to the steam generators. Although the event does not represent a complete loss, the current model assumes that no degradation of the main feedwater system is present. The partial loss of feedwater should be added to the model to account for partial losses (e.g., trip of a main feedwater pump)." Add T4P to each train of FW ANDed with a 50% split fraction.	The partial loss of feedwater was added to the model. See M%T4AA (AND of %T4A and XSPLTFA below M550 and M%T4AB (AND of %T4A and XSPLTFB) below M600. Added XSPLTFA and XSPLTFB in mutually exclusive set.
The CVCS is incapable of mitigating a SGTR as postulated in the PSA. However, successful RCS cooldown will effectively stop the leakage from a SGTR. The CVCS can then restore level and the sequence will progress as a normal plant cooldown event. The use of CVCS in this manner is credited in the PSA. However, the success criterion for the CVCS pumps is that two-of-three function. The analysis performed to support the SGTR event tree assumed maximum charging flow (3 of 3 pumps). Therefore, although possible, the current modeling is not supported by analysis. The model should be changed to either reflect the analyzed success criterion or new supporting analysis performed to support the less restrictive criterion."	Gate JTRPMPS, "NO SGTR MAKEUP FLOW FROM CHARGING PUMPS," was changed from a two-out-of-three combination gate to an OR gate.
Some modifications to the CARC system were determined to be necessary: SD-037 indicates ..."Containment Air Recirculation	MOR97 showed HVH-1, -2, and -3 in service, and HVH-4 on standby. Since the normal condition has HVH-4 running, the fail-to-start events were not applicable. The following events were removed:

MOR97 COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
Cooling (HVH-1 through HVH-4) [:]. The containment air recirculation coolers are normally in use during plant operation..."Inspection of ERFIS points VPZ9001[2,3,4]D (HVH-1,2,3,4 CV Fan Cooler Status) during January 2000 found that all 4 coolers were in service even when the ambient outside air temperature was between 30-40 F. System Engineer Ray Norris confirmed (1/31/00) that the normal system operating configuration is for all 4 coolers to be in service.	1) Gate B40D, HVH-4 COMMON CAUSE FAILURES TO START, beneath Gate B40. 2) Gate B45, HVH-4 EMERGENCY START FAILS, beneath Gate B41.
The containment fan coolers were modified by ESRs 95-764, 95-783, and 97-469 [possibly also ESR 97-000382] such that the emergency air intake butterfly valves/dampers (previously "normally-closed") were permanently opened and control air supplies were removed from them.	For air handling unit one (HVH-1), MOR97 showed event AIR FLOW THROUGH HVH-1 FAILS (Gate B13) as either from containment spray actuation failure (ES10-X) or from the damper failing to open (BAVHVH-1NN). Similar logic existed for HVH-2, HVH-3 and HVH-4. Following the modification, the only applicable failure mode is for the valve/damper to transfer closed. For HVH-1, ES10-X and BAVHVH-1NN were removed and BXVHVH-1FN (MANUAL VALVE HVH-1 TRANSFERS CLOSED / PLUGGED) was added. Gate B13 became a one-input gate. Similarly for air handling units HVH-2, HVH-3 and HVH-4, Gates ESI11X and BAVHVH-2NN below B27 were removed; Gates ES20-X and BAVHVH-3NN below B35 were removed; and Gates . ESI21X and BAHVHV-4NN below B46 were removed, respectively. Events BXVHVH-3FN, BXVHVH-3FN and BXVHVH-4FN were added. Gates B27, B35 and B46 became one-input gates. . The probability for BXVHVH-1FN, BXVHVH-2FN, BXVHVH-3FN and BXVHVH-4FN (1/24/H) was based on the 24-hour mission time and the rate used for other manual valves. Finally, Gate B99 (HVH-4 EMERGENCY RESTART FAILS) was added to provide parallel logic to that of HVH-2, HVH-3 and HVH-4.
ESR 97-00246 describes the conversion of A deepwell pump to a design similar to B and C. The common-cause failures for the deepwell pumps need to be modified to reflect this. Existing documentation indicates	With all three deepwell pumps having a similar design, common cause failures were added to the model. Operator logs for 1/1/00 - 1/31/00 were reviewed to confirm that the A pump was in service with B and C operated intermittently. Accordingly, the appropriate CCF events for failure to start (YCCFACFTS, YCCFBCFTS, YCCFABFTS, and YCCFFTS) and failure to run (YCCFACFTTR, YCCFBCFTR, YCCFABFTR and YCCFFTR) were added to the logic for B

MOR97 COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
that the normal operating configuration for the Deepwell system is for A pump to normally be in service and for B and C pumps to be used intermittently	and C deepwell pumps. See logic under Gate Y900 (placed beneath YDW3) and Y902 (YDW4). The appropriate CCF events for failure to run were added to the logic for A pump. See logic under Gate Y904 (YDW2). Probabilities for these CCF events were determined by examining the CCF probabilities for failures of the CCW pumps (another 3-pump system) and the SW pumps (a four-pump system but with exposure to raw water). All relevant CCW failure probabilities were greater than the corresponding SW failure probabilities, except for the all-pumps-fail-to-run case. Except for this case, the CCW values were adopted. See for example KCCFACFTS (1.26E-4) and KCCF%ACFTR (4.17E-4). For the all pumps fail to run case, 3 CCW pumps fail-to-run (KCCFRUN) was given as 1.74E-5, while all (4) SW pumps fail-to-run (WCCF%ABCD) was given as 1.86E-4. The probability of 3-of-4 SW pumps failing to run (e.g., WCCF%ACDFF) was given as 2.16E-4 This was conservatively adopted as the deepwell all-pumps-fail-to-run probability (YCCFFTR).
ESR 99-00037 (Change to operating the CVCS system with 2 charging pumps normally in service). ESR 99-00037 Request says, "It has recently become a recommended practice to run two charging pumps continuously. This reduces the need for packing replacement and PMs on the charging pumps. " The current model assumes one pump is normally in operation and modification is necessary to reflect the change to having two pumps in service.	Redefined XFL-JAC-NR to mean "Pumps A and C operating, B not operating," XFL-JBC-NR to mean "Pumps B and C operating, A not operating," and XFL-JAB-NR to mean "Pumps A and B operating, C not operating." Replaced J-PMPA-NR with XFL-JBC-NR. Replaced J-PMPB-NR with XFL-JAC-NR. Replaced J-PMPC-NR with XFL-JAB-NR. In initiator logic, below Gate J%PMPA1 deleted XFL-JBC-NR and added J%PUMPA9 (OR of XFL-JAB-NR, XFL-JAC-NR). Below Gate J%PUMPB1 replaced XFL-JAC-NR with J%PUMPB9 (OR of XFL-JBC-NR, XFL-JAB-NR). Below Gate J%PMPC1 replaced XFL-JAB-NR with J%PMPC9 (OR of XFL-JAC-NR, XFL-JBC-NR). Below Gate J%PMPA2, deleted J-PMPA-NR and added XFL-JBC-NR. Below Gate J%PMPB2, delete J-PMPB-NR and added XFL-JAC-NR. Below Gate J2%PMPC2, deleted J-PMPC-NR and added XFL-JAB-NR.
RLS report 96-04 indicates that 2 of 3 SG PORVs are required for rapid cooldown and depressurization during S1LOCA. This is modeled by gate #SD1 below #SD. The existing logic defines success as being any one SG PORV operating OR condenser steam dumps operating.	The condenser steam dump logic is automatically failed by a %S1 initiator input, so this was removed (gate Q001 was deleted). The remaining logic for the 3 steam generators was combined under a "two-of-three" gate, #SD1. The text label for #SD1 was changed to read, "Failure of two of three SG PORVs."

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Note: RLS report 96-04 indicates that the logic for SGTR below gate #RP1 should also require 2/3 SG PORVs operate for success. The logic below #RP1 is currently configured this way.	
Revise model to include test/maintenance events for PMGS specified by Maintenance Rule that weren't previously part of the model. Suggested modification to AFW, CCW, and EP fault trees.	Added FTMSDPTRNC (SDP Injection Train C unavailable due to T/M) below Gate FMMAF14CNN. Added FTMSDPTRNB (SDP Injection Train B unavailable due to T/M) below Gate FMMAF14BNN. Added FTMMDPTRNA (MDP Injection train A unavailable due to T/M) below Gate FMMSEGAMFN. Added FTMSDPTRNA (SDP Injection Train A unavailable due to T/M) below Gate FMMAF14ANN. Added FTMMDPTRNB (MDP Injection train B unavailable due to T/M) below Gate FMMSEGBMFN. Added FTMMDPTRNC (MDP Injection train C unavailable due to T/M) below Gate FMMSEGCMFN. Probability for the above AFW test and maintenance events (estimated at 1E-2) is consistent with other T&M events for SW booster pumps or air compressors. Added KTMCCWHXA (CCW HX Train A unavailable due to T/M), 1E-2), below Gates KMMCCWHXA and K%610. Added KTMCCWHXB (CCW HX Train B unavailable due to T/M), 1E-2), below Gates KMMCCWHXB and K%620. Probability for the above events (estimated at 1E-2) is consistent with T&M for recent CCW HX maintenance. Added NTMCB52/7 (CB 52/7 unavailable due to T/M), 2.5E-3; NTMCB52/12 (CB 52/12 Unavailable due to T/M), 2.5E-3, below Gate N17D. Added NTMCB52/19 (CB 52/19 unavailable due to T/M), 2.5E-3, and NTMCB52/20 (CB 52/20 unavailable due to T/M), 2.5E-3, below Gate NKVBUS4. Added NTMCB52/12 and NTMCB52/7 below Gate N17D-DG. Probability for the above events (estimated at 2.5E-3) is consistent with recent history of T&M for breaker outages with typical durations of one shift and no more frequent than once per year.
Modifications were made to reflect substitution of D air compressor for C, to reflect the fact that either A or B compressor can provide	See separate discussion for the instrument air system changes.

MOR97 COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
sufficient instrument air (instead of requiring both), and to correct minor errors in the model. The IA system modifications are listed in a separate discussion.	
The current EDG cooling model does not include two normally open manual valves (SW-85 and SW-89). These valves should be added for completeness. The diesel generator cooler is similar to any heat exchanger and subject to plugging or rupture. The current model does not address this failure mode. Adding a failure for the cooler should be considered.	Added PXVSW-85FN under Gate PMMEMERDGA and PXVSW-89FN under Gate PMMEMERDGB. Probability calculated using 2/14/d to be consistent with other BEs in model, such as PXSW-88FN. A diesel testing interval of 14-days was selected to be consistent with other related components. Added PHXEDGA/FN under Gate PMMEMERDGA and PHXEDGB/FN under Gate PMMEMERDGB. Probability calculated using 1/24/h to be consistent with other BEs in the model.
The fuel oil transfer system model does not include normally open manual valves FO-30A and FO-30B that are located on the day tank instrument lines. If the valves failed closed, or were plugged, instrumentation would be lost. The likelihood of this failure is low, however, it does represent a failure mode that should be in the model since it is common to all three instruments for each day tank.	Added PXVF030AFN (1/24/h) under Gate P204A and PXVF030BFN (1/24/h) under Gate P204C. Probability based on generic manual valve with same exposure as fuel oil transfer pump.
The current model does not include a CCF for EDG cooler valves TCV-1660 and TCV-1661. These valves fail open on loss of instrument air; however, they may physically fail to open. By the groundrules, a common cause failure should be included.	Added PCCFAV60/1 (2.4E-4), "CCF of pneumatic valves TCV-1660 and 1661 to open on demand," under Gates P200 and P200B. Probability set at bounding value of 10% of the component fail to open probability (see PAVT1660NN).
The current model does not include fuel oil day and main storage tank failures as is postulated for other tanks such as the RWST and CST. Failure, although unlikely, should be included in the model to be consistent. However, the exposure time should be set to	Added PTKAFOILFN, "TANK A DAY TANK CATASTROPHIC FAILURE" below Gate P204A. Probability calculated using 1/24/H because a pre-existing failure would be detected during shift rounds (8h or less) so failure during 24 h demand period is bounding. Added PTKBFOILFN, below Gate P204C. Added PTKMFOSTFN, "MAIN FUEL OIL STORAGE TANK CATASTROPHIC FAILURE" (1/24/H) below Gates P202 and P202B. Probability based on a bounding 24-hour exposure period

MOR97 COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
account for any periodic visual checks that may be more frequent than actual testing.	and generic data base for tank failures.
All four of the service water pumps are normally interconnected. If a running pump failed and its discharge check valve failed to close, it could create a short circuit effect and reduce the total delivered flow. Other PSAs have addressed the potential for pump discharge check valve failure to create pump recycle possibility and the Robinson PSA does address this event for CCW. For consistency, the model should include these failures.	The short-circuit failure mode was added to the SW system. Below Gate W800B, Gate W210F was removed and a new Gate W970 was added (3/4 SW PMPS FAIL OR 2/4 FAIL AND DISCH CHK FTC). Gate W970 is an OR-gate with inputs W210F (SW PUMPS (3/4) FAIL (NORTH HEADER) and W980 (2/4 PMPS FAIL AND DISCH CHK VLV FTC). W980 is an AND-gate with inputs W210E (SERVICE WATER PUMPS (2/4) FAIL (NORTH HDR)) and W910B (SW PUMP FAILS AND CHECK VALVE FTC). W910B is an OR-gate with 4 inputs: W911, W912, W913, and W914. W911 is an AND of W202 (SERVICE WATER PUMP A FAILS) and WCV0374AFF (CHECK VLV SW-374 FAILS TO CLOSE ON DEMAND). W912 is an AND of W204 and the B check valve, etc. Next, below W210B, W210E was removed and replaced with W920 (2/4 SW PMPS FAIL OR 1/4 SW PUMPS FAIL AND DISH CHK VLV FTC). This is an OR-gate with inputs W210E (2/4) and W910B (described above). This pattern was repeated for W210A. W210D was removed and W930 was added. W930 was an OR of W210D and W910B. Also, below Gate W800A, W210C was removed and W950 (3/4 SW PMPS FAIL OR 2/4 FAIL AND CHK VLV FTC) was added. W950 was an OR of W210C (SW PUMPS (3/4) FAIL (SOUTH HEADER)) and W960 (2/4 SW PMPS FAIL AND DISCH CHK VLV FTC). W960 was an AND of W210D and W910B.
The model appears to have omitted power supplies for Pressure Transmitters PT-1616 and PT-1684. It may be that the power supplies are accounted for elsewhere and are not required. The power requirements should be verified and any power requirements added if necessary.	Prints show power supply to both pressure switches is from 120 VAC instrument bus 1, and this was incorporated in the model. Conservatively assuming that loss of power results in loss of function. Added existing Gate C118BS1 below Gates WMMV616A and WMMV616B.
The model addressing the manual alignment of the CVCS pumps to the RWST includes a time-dependent transfer failure of manual valve CVC-358. It does not include the failure of the normally closed valve to open. This failure mode should be	Added JXV00358TF, "MANUAL VALVE CVC-358 FAILS TO OPEN." Probability (2/3/M) based on average unavailability between tests. Testing is quarterly, per OST-107. Revised probability for JXVCV358FN to be 1/24/H to reflect probability of the manual valve transferring closed or plugged during the 24-hour mission time.

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added.	
The current model for the boric acid transfer does not include failures for the boric acid tanks. Since both tanks are required, the failure of either tank is sufficient for failure and the failure of the tanks should be addressed.	Added JTKBATAAFN "TANK BAT A CATASTROPHIC FAILURE," below JBATA (1/24/h) and JTKBATBBFN "TANK BAT B CATASTROPHIC FAILURE," below JBATB (1/24/h). Probability based on 24-hour mission time and generic tank data.
%FL-6 appeared in two cases with this problem: V%FL-2 and #FL-SICS5. %FL-6 was ORed with %FL-9 into V%FL2, which was ANDed with V%FLOOD under V%FL. %FL-6 was ANDed with #FL-CONDD under #FL-SICS5.	Deleted V%FL. Deleted #FL-SICS5. Globally deleted remaining %FL-6's.
%FL-9 was under #FL-E1E2 ANDed with ~FLOOD.	Deleted #FL-E1E2. Globally deleted the remaining %FL-9's

INSTRUMENT AIR SYSTEM COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
In the Loss of Instrument Air Tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C".	Changed A%INIT3 description from "C Instrument Air Compressor..." to "D ...". Changed A%INIT301 description "C instrument air compress.." to "D..." Changed A%INIT303 description "C..." to "D..." Changed A%INIT302 description "IA compressor C..." to "IA Compressor D..." Changed A%INIT304 description "C INTR AIR..." to "D Instr. air..." Changed AFL%IACCFN description from "C..." to "D..." Changed ACP%IACCFN description from "C IA..." to "D..."
Recognize the capability of either the "A" compressor or the "B" compressor to maintain system load.	Changed A%INIT201 from "A OR B..." to "A and B...". Changed gate from OR to AND.
Modify running flag to reflect substitution of "D" compressor of the "C" compressor.	Changed XFL-CAC-NR description from "Flag C Air..." to "Flag D Air..."

INSTRUMENT AIR SYSTEM COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
In the AMM%IACCR sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C.	Changed AMM%IACCR description from "IA compressor C..." to "IA compressor D..." Changed description for AHX%IACCFN from "C compressor..." to "D compressor..." Changed description for ATK%IACCFN from "C compressor..." to "D compressor..."
In the AMM%IACCR sub-tree, reflect attendant modifications.	Changed BE ARV%3732NF to ...3829... (i.e., replace 3732 w/ 3829), change in description also. Changed BE AXV%3708FN to ...3818, change in description also. Changed BE AXV%3663FN to ...3657.., description also. Changed BE AXV%3655FN to ...3819..., description also. Changed the name for AXV%3656FN to ...3820..., description also. Probabilities for the above events remained unchanged. Added relief valves ARV%3837NF, ARV%3834NF, and ARV%3830NF. Probability (3/8760/H) based on spurious operation of a relief valve in continuous operation, duplicating existing event ARV%3829NF (was ARV%3732NF). Added dryer ATK%DRYDFN. Probability (3/8760/H) of tank failure based on the same mission time.
In the AMM%IACCR sub-tree, reflect reuse of components.	Maintained ARV%3661NF, AXV%3653FN, AXV3654FN, AXV%3660FN, and AXV%3664FN.
In the A%INIT304 sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C".	Changed description for ACPIACC%FN from "Compressor C..." to "Compressor D..." Changed description for ATMCOMPC from "Air compressor C..." to "Air compressor D..." Changed description for ACPIAC//NN from "Compressor C..." to "Compressor D..." Changed description for AOPERCOMPC from "Operator fails to restore compressor train C..." to "...train D..." Changed description for AMM%IACCS from "IA compressor C..." to "IA compressor D..."

In the A%INIT304 sub-tree, reflect attendant modifications.	Added ATKDRYD%FN (Description: Tank D IA receiver catastrophic failure), event data: c/factor/units=1/72/h. .
	Probability of tank failure based on the same mission time as other components in train (see AHXIACC%FN).
	Added ARV3837%NF. Description, "Relief valve 3837 spurious operation," 1/72/h.
	Added ARV3834%NF, "3834".
	Added ARV3830%NF, "3830".
	Probability of spurious operation based on this same 72-hour mission time.

In the A%INIT304 sub-tree, reflect reuse of component.	Maintain OP%21.

INSTRUMENT AIR SYSTEM COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
In the AMM%IACCS sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C" and attendant modifications.	Changed description for AFLIACC%FN from "C..." to "D...." Changed description for AHXIACC%FN from "C..." to "D..." Changed description for ATKIACC%FN from "C..." to" D..."
In the AMM%IACCS sub-tree, reflect attendant modifications.	For ARV3732%NF, changed BE and desc to ...3829... For AXV3708%FN, changed BE and desc to ...3818... For AXV3663%FN, changed BE and desc to ...3657 For AXV3655%FN, changed BE and desc to ...3819... For AXV3656%FN, changed BE and desc to ...3820... Probabilities for the above events remained unchanged.
In the AMM%IACCS sub-tree, reflect reuse of components.	Maintained ACVI3666NN, ARV3661%NF, AXV3653%FN, AXV3654%FN, AXV3660%FN and AXVI3664FN.
Include tank failure for IA dryer.	Below AMM%PACR, added BE ATK%DRYPFN (Tank PAC IA dryer catastrophic failure, 3/8760/h). Probability based on mission time for other components in train and generic failure rate. Below AMM%PACS, added BE ATKDRYP%FN (Tank PAC IA dryer catastrophic failure, 1/432/h). Probability based on mission time for other components in train and generic failure rate.
In the LOSS OF INSTRUMENT AIR SUPPLY TO SG PORVS OR MSIVS sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C" and attendant modifications.	Changed description of Gate A223A, to "Loss of air from IA air compressor trains A & B & D." Changed description for AIATRNS to "Air compressor trains A & B & D fail" Changed description for AIATRNS1 to "Air compressor trains A & B & D fail"
In the AIATRNS1 sub-tree (under A223 sub-tree), recognize the capability of either the "A" compressor or the "B" compressor to maintain system load.	Changed description for Gate A224 from "A or B..." to "A and B..." Changed Gate A224 from an OR-gate to an AND-gate.
In the AIATRNS1 sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C".	Changed description for Gate A501 from "Loss of air from C..." to "Loss of air from D..."
In the A501 sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C"	Changed description for Gate A-CIAC-OOS from "C Instrument..." to "D Instrument..." Changed description for Gate AMMIA/C from "IA compressor C..." to "IA compressor D..."
In the A501 sub-tree, reflect attendant modifications.	Added ARVI3837NF ("relief valve 3837 spurious operation," 1/24/h). Added " ARVI3834NF " Added " ARVI3830NF "

INSTRUMENT AIR SYSTEM COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
	Probabilities based on exposure time for other components in train and generic failure rate. Added ATKADRYDFN, ("Tank D IA dryer catastrophic failure," 1/24/h). Probabilities based on exposure time for other components in train and generic failure rate.
In the A501 sub-tree, reflect reuse of components.	Maintained ASTS1850NF, NMCC13, AXVIA18/FN, and AOPERCOMPC.
In the AMMIA/C sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C"	Changed description for AFLIAC//FN to "D compressor..." Changed description for AHXIAC//FN to "D compressor..." Changed description for ACPIAC//FN to "Compressor D..." Changed description for ATKIA//CFN to "D compressor..."
In the AMMIA/C sub-tree, reflect attendant modifications.	Replaced BE AXVI3708FN with AXVI3818FN. Replaced BE AXVI3655FN with AXVI3819FN. Replaced BE AXVI3656FN with AXVI3820FN. Replaced BE AXVI3663FN with AXVI3657FN. Replaced BE ARVI3732NF with ARVI3829NF. Probabilities for the above events remained unchanged.
In the AMMAI/C sub-tree, reflect reuse of components.	Maintained AXVI3653FN, ACVI3666NN, AXVI3654FN, AXVI3660FN, AXVI3664FN, and ARVI361NF.
In the A-CIAC-OOS sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C".	Changed A-CAC-OOS2 description from "C instrument air..." to "D instrument air..."
In the A-CAC-OOS2 sub-tree, reflect substitution of Instrument Air Compressor "D" for Instrument Air Compressor "C".	ATMCOMPC, changed desc to "Air compressor train D..." ACPIAC//NN, changed desc to "Compressor C..." to "Compressor D..."
In the A-CAC-OOS2 sub-tree, reflect attendant modifications.	Added BE ACVA3836NN, CV fails to open, 1/1/N. Added BE ACVA3666NN, CV fails to open, 1/1/N. Probabilities based on same parameters as existing event in tree (see ACPIAC//FN).
In the A-CAC-OOS2 sub-tree, reflect reuse of components.	Maintained OPER-21.
In the A700 sub-tree, reflect attendant modifications.	Added AXVI3664FN, Manual valve 3664 transfers closed, 2/18/M. Added AXVI3663FN, Check valve 3663 transfers closed, 2/18/M. Added AXVI3662FN, Check valve 3662FN transfers closed, 2/18/M. The above valves are in crosstie, which allows primary AC to parallel D. As these valves are not routinely demanded and valve position is assumed to be verified only during line-ups on startup after outage, probabilities assume an 18-month test interval. Added ACVI3660FN, ...3660. Probability for this check valve based on the above.

INSTRUMENT AIR SYSTEM COMMENT AND INVESTIGATION	MOR99\RNPSYSTM.CAF ACTION TAKEN
In the AMMPRISW sub-tree, reflect attendant modifications.	Added AXVSW608FN, "Manual valve 608 transfers closed," 1/24/H. Probability for this event based on same values for other manual valves in train.
In the A-PAC-OOS2 sub-tree, reflect attendant modifications.	Added ACVIA495NN, "Check valve 495 fails to open," 1/1/N. Probability for this event based on same values for other fail-to-open valves.
In the AMMSGPORV sub-tree, reflect attendant modifications.	Added AXVIA298FN, Manual valve transfers closed, 2/6/M. Probability for this event based on same values for other valves in train (see AXVIA297AFN).
In the AMMSGPN2 sub-tree, add missing information.	AMMSGPN2, added description, "Nitrogen line faults."
In the AMMSGPN2 sub-tree, reflect attendant modifications	Added AXVIA299FN, "Manual valve 299 transfers closed", (1/24/h). Probability for this event based on the 24-hour mission time and generic values for transfer closed. Changed AXVSDN38TF description to "Manual valve (Standby) SDN-28 fails to open."
In the ASGPORV-N2 sub-tree, reflect attendant modifications	Added ATKN2ACCFN, "Tank N2 accumulator catastrophic failure," 1/24/h. Probability for this event based on the 24-hour mission time and generic values for tank failure.
In the A710 sub-tree, reflect attendant modifications	Below AMM1093A, deleted AAVFEA//FN. Below AMM1093B, deleted AAVFEB//FN. Below AMM1093C, deleted AAVFEC//FN.
In the R2000 sub-tree, reflect attendant modifications.	Below AMMN2-456, deleted ACVOPP09NN. Below AMMN2-455C, deleted ACVOPP10NN.
In the AMMSDN1727 sub-tree, reflect attendant modifications.	Changed description for AXVSDN30FN to "Manual valve SDN-20..."

Functional Logic Changes

The following changes were applied to RNPFNLOG.CAF.

MOR97\RNPFNLOG.CAF COMMENT AND INVESTIGATION	MOR99\RNPFNLOG.CAF ACTION TAKEN
Add ~TQDGX below #T-LOCA and ~FQDGX below #F-LOCA. Same reasoning as S1LOCA.	~TQDGX added below #T-LOCA. ~FQDGX added below #F-LOCA. Considered adding below #L-LOCAS and rejected, analysis showed CST inventory adequate for LOOP sequences with no reactor coolant pump energy input.
The S2 LOCA was restored to the model following the update to the LOCA recirculation procedures that allows for low-pressure recirculation and does not require high-pressure injection in all cases. For the IPE, this	In the S2 model, event Z consists of HHSI (H999I) "ORed" with OPER-7 (an additional operator failure to align recirc). HHSI is already modeled in event "U" and operator failure to align recirc is modeled by OPER-1

MOR97\RNPFNLOG.CAF COMMENT AND INVESTIGATION	MOR99\RNPFNLOG.CAF ACTION TAKEN
requirement removed any difference between the S2 LOCA and the medium LOCA classes and the two were grouped. The RCS pressure at the time of recirculation for the medium LOCA is sufficiently low, however, to now credit low-pressure recirculation. This improves the likelihood of recirculation success by removing the requirement to align the RHR pumps to the safety injection pumps. It was specifically added for the large LOCA event tree to account for the phased mission. The initial success criterion required one of two RHR pumps for injection until they were stopped to swap sources. At that point, the safety injection pumps were required to maintain adequate flow. All pumps were then stopped, safety injection was aligned to the RHR pump discharge, and all pumps were then restarted. Thus, changes in the success criterion over the mission occurred and Event Z was needed to account for these changes. For the S2 LOCA and medium LOCA cases, safety injection is the required makeup source and RHR injection is not addressed. Recirculation is then provided using either high-pressure recirculation (for S2 LOCA) or low-pressure recirculation (for medium LOCA). In either case, however, there is no change in the injection success criterion. Event Z is modeled in both event trees and may result in some double counting of safety injection failures. It is recommended that both event trees be modified to remove Event Z and that any operator actions associated with recirculation be moved to Event X. Event X would account for any restarting of the safety injection pumps and the actions to accomplish recirculation alignment, as is the case for the transient and S1 LOCA event trees. Event Z should be retained in the large LOCA event tree."	under event X. The HHSI logic under "Z" is redundant and the contribution from OPER-7 is either redundant or at least negligible. Event Z was deleted from the S2 top logic ["~S2Z" was deleted from #S2LOCAS in rnpfnlog.caf]. The event tree graphic and description for S2 will have to be corrected. The same circumstances apply to the medium local top logic and event Z was deleted there, too. ["~MZ" was deleted from #MEDIUM in rnpfnlog.caf].
Comparison of S1 event tree with top logic showed that sequence ~sdgx was not modeled.	This sequence was added below #S1LOCAS (A gate ~sdgx was added to the existing AND gate [in RNPFNLOG.CAF].) Same change made to TQ and F sequences.

The revised RNPSYTM.CAF (Attachment W), the revised RNPFNLOG.CAF (Attachment N) and the pre-existing RNPTPLOG.CAF (Attachment X) were then merged to form MOR99\RNP.CAF (Attachment E). Printouts of these files are included as Attachments W, N, X and E, respectively.

<u>Database Changes</u>

The following changes were applied to RNP.BE:

COMMENT AND INVESTIGATION	MOR99\RNP.BE ACTION TAKEN
Correct data for NCB5226AFN to show 1/24/h as described in MOR calc.	Correction made. See A/R 13608.
Update internal flooding frequencies	Done, as documented herein.
Update ISLOCA frequency	Done, as documented herein.
Update combination operator action values for revised OPER-37 value.	Done, as documented in calculation RNP-F/PSA-0008.
Correct value for X-OQ-0333 to be 1.50E-03.	Done, as documented in calculation RNP-F/PSA-0008.
Correct value for X-OT-0277 to be 1.50E-03.	Done, as documented in calculation RNP-F/PSA-0008.
Revise name for "X-OM-004" to be "X-OM-0004"	Done, as documented in calculation RNP-F/PSA-0008.
Add the following description for OPER-8, "OPERATOR FAILS TO MANUALY ISOLATE SW TO TURBINE BLDG".	Done, as documented in calculation RNP-F/PSA-0008.
The following operator actions are listed in MOR97\RNP.BE but do not appear in MOR97\RNP.CAF and may be purged: OPER-13, OPER-18C, OPER-18D, OPER-22, OPER-CSS, OPER-VOLT. (The following operator actions also do not appear in RNP.CAF but should be maintained for future reference: OPER-17, OPER-25.)	Done, as documented in calculation RNP-F/PSA-0008.

An update to the internal flooding event assessment was incorporated to examine the IPE analysis. The existing analyses of initiating event frequency were updated.

This report was prepared prior to the determination that PSA work would be prepared under in accordance with engineering procedures and applicable portions of CP&L's Appendix B Quality Assurance program and would be maintained under configuration control. Accordingly, the report was "grandfathered" (meaning it has been accepted for use, "as is").

The following initiators were revised as documented in Reference 2.3.

INITIATOR	VALUE	COMMENT
%FL-1	1.90E-05	Revise definition to definition for old FL-7
%FL-2	7.50E-05	Revise definition to definition for old FL-8
%FL-3	6.20E-05	
%FL-4	6.40E-05	
%FL-5	3.60E-06	
%FL-6	N/A	Delete - no longer used
%FL-7	N/A	Delete - no longer used
%FL-8	N/A	Delete - no longer used
%FL-9	N/A	Delete - no longer used
%FBAT	N/A	Delete - no longer used
%FREL	N/A	Delete - no longer used

Additional changes to flood logic were required to remove flags and split fractions AND-ed with deleted values. All changes described were applied to RNPSYSTM.CAF, as described above.

An update to the interfacing system loss of coolant accident (ISLOCA) was incorporated to examine the IPE analysis in additional detail. The existing analyses of initiating event frequency and accident sequence progression were updated and the potential for operator intervention considered. Analyses performed subsequent to the IPE on selected inputs to the analysis of RHR hot leg suction ISLOCA were also incorporated to provide a complete analysis.

This report was prepared, with one exception, prior to the determination that PSA work would be prepared under in accordance with engineering procedures and applicable portions of CP&L's Appendix B Quality Assurance program and would be maintained under configuration control. Accordingly, the report was "grandfathered" (meaning it has been accepted for use, "as is"). The one exception is the reanalysis of initiating event frequency to account for the RNP RHR configuration wherein RHR-751 has power removed. Therefore, the calculation documenting this work is limited to the review of this portion of the report. The following frequency was revised, as documented in Reference 2.4.

INITIATOR	VALUE	COMMENT
%ISLOCA	1.34E-06	None.

The above changes were incorporated and the file saved as RNP.BE in the MOR99 directory. Printouts of the database, and the associated .GT and .TC files are included as Attachments D, F and G, respectively.

File RNPLERF2.BE was created by merging MOR99\RNP.BE and RNPLERF.BE (unchanged from MOR97). All operator actions in the merged file were then set to 1.0. and the file was saved as RNPLERF2.BE. This file will be used to load LERF split fractions in a subsequent step. The contents of this file are not printed and attached hereto. A printout of RNPLERF.BE is included as Attachment O. The associated .GT and .TC files are empty and are printed.

Mutually Exclusive File Changes

COMMENT AND INVESTIGATION	MOR99\RNPMUTEX.CUT ACTION TAKEN
A partial loss of feedwater event to both trains is the loss of feedwater event and therefore should be excluded.	Added XSPLTFA and XSPLTFB to MOR97 mutually exclusive file.
Both trains of CCW would not be scheduled for routine test and maintenance at the same time.	Added KTMCCWHXA and KTMCCWHXB to MOR97 mutually exclusive file.

The above changes were incorporated into the MOR97 mutually exclusive file and saved as MOR99\RNPMUTEX.CUT. A printout of this file is included as Attachment Q.

Rule-Based Recovery File Changes

A minor revision to the text-based recovery rule file was prepared to revise the assumed value for operator action OPER-37, to correct several minor typographical errors and to and remove references to modules. This file is referred to as L1_RUL99.TXT and is documented in Reference 2.5. A printout of the recovery file is not included herein.

Level 1 Quantification

MOR99 was solved (quantified) for Level 1 using the following steps:

1. Obtained RNP.CAF (as described above).
2. Obtained RNP.BE (as described above).
3. Obtained RNPMUTEX.CUT (as described above).
4. Obtained L1_RUL99.TXT (as described above).
5. Obtained RNPFLAGS.CAF (unchanged from MOR97). A printout of this file is included as Attachment M.
6. Obtained RNPSEQ.CAF (unchanged from MOR97). A printout of this file is included as Attachment U.
7. Used PRAQUANT (file RNPCDF.QNT) to develop the Level 1 cutset file (RNPCDF.CUT). Truncation was set at 4 orders of magnitude below the

result, or 4.00E-09. A printout of the specific configuration is included as Attachment I.

8. Attached sequence recovery labels to RNPCDF.CUT by merging RNP.CAF and RNPFLAGS.CAF; setting .T to "True" and .F to "False"; compressing True/False; merging RNPSEQ.CAF; and saving the resulting file as RNPRSEQ.CAF. A printout of RNPRSEQ.CAF is included as Attachment T.
9. Used QRECOVER to add sequence labels to the file RNPCDF.CUT, using the above RNPRSEQ.CAF as the Recovery Rule File. A printout of the resulting cutset file is included as Attachment H.

<u>CDF Importance Measure Report</u>

The importance report was generated by sorting RNPCDF.CUT by probability. A printout of this report is also included with Attachment H.

<u>Level 2 Quantification</u>

The "delete term" method was used to quantify the eight cutset files used in Level 2 quantification. The following steps were followed:

1. Obtained files RNP.CAF, RNPLERF2.BE, RNPMUTEX.CUT, L1_RUL99.TXT, RNPFLAGS.CAF, RNPPDS.CAF (unchanged from MOR97). A printout of RNPPDS.CAF is included as Attachment R.
2. Used PRAQUANT (file RNPCNMT.QNT) to develop the Level 2 cutset files. Truncation was set at 4 orders of magnitude below the result, or 4.00E-09. A printout of the specific configuration is included as Attachment K. The following eight events comprised the Level 2 model:
   - #CDNPD3
   - #CDNPD2
   - #CDNPD1
   - #CDMPD3
   - #CDMPD2
   - #CDMPD1
   - #CDMGNM
   - #CDMGMIN
3. Appended the eight cutsets created above into RNPCNMT.CUT.
4. Used QRECOVER to add sequence labels to RNPCNMT.CUT, using the above RNPRSEQ.CAF as the Recovery Rule File.
5. Attached plant damage state labels to RNPCNMT.CUT by merging RNP.CAF and RNPFLAGS.CAF; setting .T to "True" and .F to "False"; compressing True/False; merging RNPPDS.CAF and saving the resulting file as RNPRPDS.CAF. A printout of RNPRPDS.CAF is included as Attachment S.

6. Used QRECOVER a second time to add plant damage state labels to RNPCNMT.CUT, using the above RNPRPDS.CAF as the Recovery Rule File. Note that the QRECOVER option was set to allow for two recoveries.
7. Created CNMT and LERF modules within RNPCNMT.CUT using Cutset Editor function: Edit-Module Operations.
8. Replaced PDS label probabilities in the RNPCNMT.CUT LERF module with the LERF split fractions by loading new database probabilities from RNPLERF2.BE. A printout of the resulting cutset file is included as Attachment J.

CNMT and LERF Importance Measure Reports

The CNMT Importance Measure Report and the LERF Importance Measure Report were generated by sorting the CNMT and LERF modules of RNPCNMT.CUT by probability. Printouts of these reports are also included with Attachment J.

Other Miscellaneous Files

Text files RNPCOMP.SEN and RNPSYS.SEN were unchanged from MOR97 and remain applicable for MOR99. Printouts for these files are included as Attachments L and V, respectively.

As modules are no longer used in RNPSYSTM.CAF and RNP.CAF, the RNPMOD.CUT is no longer used. Attachment P was used for the previous model and is empty for MOR99.

4.4 Precautions and Limitations

4.4.1 The Model is under the purview of the PSA Unit. Use of the Model by other organizations should be with the full knowledge of the PSA Unit.

4.4.2 This calculation documents a revision to the fault tree to reflect a plant modification to the instrument air compressors, in particular to replace IA compressor "C" with IA compressor "D". Gate descriptions have been changed to reflect this plant modification; however, in many cases the gate names themselves have not been changed. For example, the description, "C INSTRUMENT AIR COMPRESSOR UNAVAILABLE" has been revised to "D INSTRUMENT AIR COMPRESSOR UNAVAILABLE" while the OR gate name "A-CAC-OOS2" remains unchanged.

4.4.3 As discussed above, this calculation documents a revision to the fault tree to reflect a plant modification to the instrument air compressors. The pre-existing Human Reliability Assessment (Appendix E) documentation for OPER-21 refers to steps in procedure AOP-017 and to Instrument Air Compressor "C". Procedure AOP-017 has been revised to refer to IA Compressor "D". The HRA documentation should be updated to be consistent with the updated fault tree (MOR99) and the current procedure.

4.4.4 A/R 13724 directs that the database be purged to delete unused top logic in RNP.CAF. As the unused logic had no impact on MOR and due to resource limitations, this improvement item was postponed.

When performing applications, care should be taken to ensure that model changes propagate to the tree top of interest.

5.0 CONCLUSIONS

The attached files document the RNP Updated IPE PSA Model. Use of the Model by other organizations should be with the full knowledge of the PSA Unit.

6.0 CROSS DISCIPLINE IMPACT

This calculation has no impact on any design documents outside of the PSA Unit of NFM&SA. Therefore, no additional review is required.

7.0 LICENSING IMPACT

The IPE was developed in response to CP&L's commitment to Generic Letter 88-20. This model was intended to be updated and maintained as a "living document" for use as a risk management tool. As this intent was not a licensing commitment, this calculation has no impact on any licensing document (i.e., the IPE). The Updated IPE will not be submitted; however, other licensing actions such as LER responses may reference the Updated Model.

This calculation has no impact on any licensing documents. Therefore, no additional review is required.

8.0 SCOPE OF REVIEW

The following is the suggested minimum scope for this review:

1. Confirm that model changes were implemented as shown in the tables in Section 4.
2. Complete the EGR-NGGC-0003 Record of Lead Review (Engineering Review) and include as Attachment AA.