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March 9, 2000

2CAN030003

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station OP1-17
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Proposed License Change For Cycle 14 Risk-Informed Operation

Gentlemen:

On February 11, 2000 (2CAN020005), the operational assessment of steam generator tubing for the remainder of cycle 14 was submitted to the Staff. The emphasis of the assessment was to evaluate approximately one half-cycle operation for eggcrate axial flaws on the hot leg portion of the steam generators (SGs). The assessment utilized full cycle data for deterministic evaluations of different damage mechanisms.

The operational assessment determined an acceptable runtime of 0.83 effective full power year (EFPY), which bounds the actual run time of approximately 0.80 EFPY until the next scheduled ANO-2 refueling outage (2R14) in September 2000. The ANO-2 SGs will be replaced during the 2R14 outage.

To provide an additional demonstration that ANO-2 is safe to operate until the 2R14 outage, a risk-informed analysis of the eggcrate axial flaws has been completed. To utilize this analysis, a license amendment to permit use of a risk-informed determination for this specific damage mechanism has been developed. The license amendment would be applicable for the remainder of the current operating cycle which ends in September 2000 at the start of the 2R14 outage. Steam generator tube integrity assessments for cycle 15 and beyond will revert to the current deterministic licensing basis.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

Entergy Operations requests that the effective date for this change be upon NRC issuance. Although this request is neither exigent nor emergency, your prompt review is requested.

A001

Very truly yours,

CGA/jjd
attachments

To the best of my knowledge and belief, the statements contained in this submittal are true.

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for Pope **County and the State of Arkansas, this** 9th **day of** March**, 2000.**

Notary Public
My Commission Expires 12/15/2007

"OFFICIAL SEAL"
Andrea Pierce
Notary Public, State of Arkansas
County of Pope
My Commission Exp. 12/15/2007

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ATTACHMENT

TO

2CAN030003

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT TWO

DOCKET NO. 50-368

DESCRIPTION OF PROPOSED CHANGE

The proposed change to the Arkansas Nuclear One, Unit 2 (ANO-2), operating license allows operation for the remainder of cycle 14 based, in part, on a risk-informed analysis of steam generator (SG) tube integrity consistent with Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant–Specific Changes to the Licensing Basis." New license condition 2.C.(10) is added as follows:

> "For Cycle 14 only, Entergy Operations shall be permitted to operate the reactor based on a risk-informed demonstration that predicted steam generator tube integrity, with consideration of eggcrate axial flaws, is adequate to meet Regulatory Guide 1.174 numerical acceptance criteria. In accordance with Principle 5 in Regulatory Guide 1.174 concerning monitoring operational experience to ensure that performance is consistent with risk predictions, if Entergy Operations plugs or repairs steam generator tubes during Cycle 14, then the steam generators shall be reinspected to the extent necessary to verify that they have been returned to a condition consistent with the risk assessment."

BACKGROUND

The steam generator tubing inspection during the planned mid-cycle outage 2P99 in September 1999, focused on the lower eggcrates on the hot leg side of both generators. Six indications identified during the inspection were chosen for in-situ pressure testing to confirm that the three times the normal operating differential pressure (3ΔP) performance criterion for SG tubing structural integrity was met. All six tubes met the design basis accident structural and leakage requirements. None of the six tested tubes leaked at or below the main steam line break (MSLB) pressure of 2500 psi. One of the six tubes tested did fail to achieve the target pressure of 4650 psig (72-72). The tube was not pressurized to 4650 psig due to the inability of the pump to maintain a high enough pressure while maintaining an elevated flow rate.

Since it was initially inconclusive whether Tube 72-72 met the 3ΔP margin criterion, the initial operational assessment developed to support initial plant operation after the outage utilized an assumption of tube burst. The assessment, submitted December 21, 1999 (2CAN129911), determined an operating runtime of 7.0 effective full power months (EFPM) to be the point at which the worse case flaw would exceed the 3ΔP criterion. The 7.0 EFPM runtime would correspond to a June 25, 2000, plant shutdown.

On February 11, 2000 (2CAN020005), the finalized operational assessment for the remainder of cycle 14 was submitted. The emphasis of this assessment was to evaluate approximately one half-cycle operation for the axial cracks on the hot leg portion of the SG. The final assessment utilized full cycle data based on a deterministic evaluation for all mechanisms. This evaluation followed guidance provided by Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," for performing condition monitoring and operational assessments of steam generator tubing degradation. Additionally, guidance from the Electric

Power Research Institute (EPRI) "Steam Generator Integrity Assessment Guidelines" was also used.

An input into the final operational assessment was an evaluation performed to determine the burst pressure of tube 72-72. Based upon the result of post in-situ test eddy current and studies of burst test of electric discharge machine (EDM) notched samples, it was determined that tube 72-72 met 3ΔP. Therefore, it was concluded that all the indications in-situ pressure tested during 2P99 met the 3ΔP structural criterion. Details of this evaluation were included in the final operational assessment.

For the bounding damage mechanism, eggcrate axial flaws, the final operational assessment determined an acceptable runtime of 0.83 effective full power year (EFPY), which bounds the actual run time of approximately 0.80 EFPY until the next scheduled ANO-2 refueling outage (2R14) in September 2000. The ANO-2 SGs will be replaced during the 2R14 outage.

Entergy Operations believes the deterministic results documented in the final operational assessment to be an adequate basis for operation until 2R14. However, to provide an additional demonstration that ANO-2 is safe to operate until the 2R14 outage, a risk-informed analysis of the eggcrate axial flaws has been completed. To utilize this analysis, a license amendment to permit use of a risk-informed determination for this specific damage mechanism, in lieu of the deterministic analysis included in the current licensing basis, has been developed. The license amendment would be applicable for the remainder of the current operating cycle which ends in September 2000 at the start of the 2R14 outage. Steam generator tube integrity assessments for cycle 15 and beyond will revert to the current deterministic licensing basis.

DISCUSSION OF CHANGE

Energy Operations has developed the attached severe accident risk assessment to complement the deterministic steam generator tube operational assessment. The risk assessment considered accident sequences affected by this change that are contributors to the core damage frequency (CDF) currently assumed in the ANO-2 probabilistic safety assessment (PSA). Since many of the sequences are beyond the Safety Analysis Report accident analysis assumptions (current licensing basis), they are given the term severe accidents. The methods used to perform the risk assessment are similar to the approach used in NUREG 1570, "Risk-Assessment of Severe Accident-Induced Steam Generator Tube Rupture."

For the proposed change, the risk assessment assumed a plant shutdown and steam generator tubing inspection beginning on May 15, 2000. The CDF and large early release fraction (LERF) were calculated for two operational scenarios: (1) operation with the proposed May inspection; and (2) operation without the May inspection. The difference in the CDF and LERF results between these two scenarios were then determined and compared with the RG 1.174 acceptance criteria. It should be noted that the assumption of a May 15, 2000, shutdown for the analysis is conservative since the deterministic operational assessment submitted previously concluded that margin existed with a continuous run to 2R14, and even the preliminary assessment submitted in December 1999, concluded that operation until late June 2000, was justified.

The base line CDF for ANO-2 has been calculated at 1.97E-5/rx-yr, while the baseline LERF is 4.81E-6/rx-yr. The ΔCDF between the May 15, 2000, outage and the no outage scenarios is less than 1E-7/rx-yr, which is in Region III of Figure 4.0 of RG 1.174, and thus considered an acceptable very small change. The ΔLERF has been calculated in several ways. The most conservative analysis performed yielded a ΔLERF of 3.8E-7/rx-yr, which is within Region II of Figure 4.0 of RG 1.174. This ΔLERF is characterized as a small change per the RG. Therefore, the difference in risk between performing an additional steam generator tubing inspection in May 2000 and operating without another inspection until shutdown for 2R14 is considered small and acceptable per RG 1.174.

RG 1.174 provides five principles for risk-informed decision making. Each of these principles is addressed below for the proposed change to the ANO-2 licensing basis.

Principle 1 The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

General Design Criteria 14 states in part that the reactor coolant pressure boundary shall be tested so as to have an extremely low probability of gross rupture. The attached analysis shows that the probability of tube burst for axial eggcrate SG tube flaws at the analyzed main steam line break pressure of 2500 psi will increase approximately 0.1 percent as a result of the proposed change. The probability of having a spontaneous SG tube rupture (at the normal operating differential pressure of 1350 psi) is expected to remain essentially constant whether or not an additional inspection is performed.

Principle 2 The proposed change is consistent with the defense in depth philosophy.

Because the proposed change involves the integrity of SG tubes, it affects two of the three physical barriers provided to prevent the release of radioactive material to the environment. Because of this, the LERF criterion addressed by Principle 4 should be the primary consideration for determining the adequacy of these barriers. From a probabilistic perspective, defense in depth is provided by the combination of challenge frequency and conditional probability of failure due to the challenge. Without an additional inspection, the probability of the steam generator tubes retaining sufficient structural integrity to survive the design basis main steam line break is > 99%.

The most limiting severe accident (i.e., the high reactor coolant system (RCS) pressure/dry SG/low SG pressure condition) will likely result in a temperature-induced SG tube rupture regardless of whether or not an additional SG inspection is performed prior to 2R14. Given this assumption, the primary emphasis is in reduction of the challenge frequency for conditions which may lead to a temperature-induced SG tube rupture event. Changes to both the ANO-2 emergency operating procedures and the ANO-2 severe accident management guidelines have been made to provide assurance that the pressure difference across the SG tubes is minimized and that the reactor coolant pump loop seal remains intact.

Additionally, the operational assessment submitted on February 11, 2000, documents deterministically that the steam generator tubing structural integrity criterion of 3ΔP will be maintained to the 2R14 outage with margin.

Principle 3 The proposed change maintains sufficient safety margins.

Based on the attached analysis of the effect of not performing an additional inspection in May 2000, there is an 91% probability of meeting the 3ΔP burst criterion (4050 psi) throughout the current cycle. This analysis indicates that the probability of maintaining sufficient safety margins will remain adequate.

Additionally, the six worst flaws identified during the last inspection were in-situ pressure tested. All six flaws were successfully tested to 1.43 times the main steam line break accident differential pressure (3575 psi) without burst or leak. The pressure testing demonstrated that five of the six flaws also passed the 3ΔP burst criterion (4050 psi), with the sixth calculated to have met the criterion.

Principle 4 When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

The change is calculated to result in a small to very small increase in the probability of a severe accident induced SG tube rupture event. The projected ΔCDF for this change is very small and falls within Region III of the RG 1.174 acceptance criteria. Using a conservative interpretation of the RG 1.174 definition of ΔLERF, the change is expected to result in a small increase to the ANO-2 LERF (i.e., the ΔLERF falls within Region II). In addition, this increase will apply only for a short interval of time (until the September 2000 refueling outage, 2R14). During 2R14, the currently installed SGs will be replaced. Thus, these risk increases associated with the change are acceptably small and meet the intent of the policy statement.

Principle 5 The impact of the proposed change should be monitored using performance measurement strategies.

The purpose of this principle is to prevent repetitive occurrence of undesirable conditions by detecting and correcting conditions that are not consistent with the assumptions in the risk assessment and other analyses used to support the change. Recurrence of the current steam generator tube condition is precluded after the end of cycle 14 since the SGs are being replaced. However, to address the potential for unexpected SG tube leakage or rupture during the remainder of cycle 14, the proposed license condition requires inspections and reanalysis of risk predictions sufficient to reestablish conformance with RG 1.174 guidelines should ANO-2 be shutdown due to a primary-to-secondary leakage through the SGs prior to 2R14.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. is proposing that the Arkansas Nuclear One, Unit 2 (ANO-2), Operating License be amended to allow continued cycle 14 operation based on a risk-informed approach as one means to evaluate steam generator tube structural integrity due to eggcrate axial flaws. The risk-informed approach is utilized to determine the magnitude of the change in risk for proposed operation for a portion of cycle 14 beyond that analyzed in a traditional deterministic manner. While the maximum differential pressure that a steam generator tube experiences during a design basis event is 2500 psi, this change evaluates the effects of a minor reduction in the steam generator tubing structural integrity margin of safety (4050 psi).

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

A steam generator tube rupture is an accident previously evaluated in the ANO-2 Safety Analysis Report. The probability of tube burst under design basis accident conditions is only slightly increased by the proposed change due to the minor reduction in margin of safety associated with tubing structural integrity, but is within the current industry guidance of NEI 97-06, "Steam Generator Program Guidelines." Detailed studies have been performed to evaluate the probable condition of the steam generator tubing for the remainder of cycle 14 operation. These studies show less than a 0.1 percent increase in the probability of tube rupture under worst case design basis accident conditions as a result of the proposed change.

This change does not modify any parameter that will increase radioactivity in the primary system or increase the amount of radioactive steam released from the secondary safety valves or atmospheric dump valves in the event of a tube rupture.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The scope of this change does not establish a potential new accident precursor. The design basis accident analyses for ANO-2 include the consequences of a double-ended break of one steam generator tube which bounds other postulated failure mechanisms. The proposed change does not modify any mode of operation or modify existing periodic inservice inspection requirements.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed change justifies a minor reduction in the steam generator tubing structural integrity margin of safety of three times normal differential operating pressure (4050 psi). However, the margin of safety for a tube burst still remains well in excess of the 2500 psi maximum differential pressure used in the design basis accident analysis for a main steam line break. The proposed change is technically consistent with the criteria of NEI 97-06 and Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis".

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

10 CFR 51.22(c) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant

increase in the amounts of any effluents that may be released off-site, or (3) result in a significant increase in individual or cumulative occupational radiation exposure. Entergy Operations, Inc. has reviewed this license amendment and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because this change does not modify methods of operation, maintenance, or inspection or increase occupational source terms.

PROPOSED LICENSE CHANGE

2.C.(4) (Number has never been used.)

(5) EOI shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following.

1. Provisions establishing preventative maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(6) EOI shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

2.C.(7) Deleted per Amendment 78, 7/22/86.

(8) Antitrust Conditions

EOI shall not market or broker power or energy from Arkansas Nuclear One, Unit 2. Entergy Arkansas, Inc. is responsible and accountable for the actions of its agents to the extent said agent's actions affect the marketing or brokering of power or energy from ANO, Unit 2.

(9) Rod Average Fuel Burnup

Entergy Operations is authorized to operate the facility with an individual rod average fuel burnup (burnup averaged over the length of a fuel rod) not to exceed 60 megawatt-days/kilogram of uranium.

(10) Cycle 14 Risk-Informed Operation

For Cycle 14 only, Entergy Operations shall be permitted to operate the reactor based on a risk-informed demonstration that predicted steam generator tube integrity, with consideration of eggcrate axial flaws, is adequate to meet Regulatory Guide 1.174 numerical acceptance criteria. In accordance with Principle 5 in Regulatory Guide 1.174 concerning monitoring operational experience to ensure that performance is consistent with risk predictions, if Entergy Operations plugs or repairs steam generator tubes during Cycle 14, then the steam generators shall be reinspected to the extent necessary to verify that they have been returned to a condition consistent with the risk assessment.

D. Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Industrial Security Plan," with revisions submitted through August 4, 1995. The Industrial Security Plan also includes the requirements for guard training and qualification in Appendix A of the safeguards contingency events in Chapter 7. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

E. This license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, EOI will prepare and record an environmental evaluation for such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated, in the Final Environmental Statement (NUREG-0254) or any addendum thereto, EOI shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation.

F. This license is effective as of the date of issuance and shall expire at midnight, July 17, 2018.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by D. B. Vassallo for

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:
Preoperational Tests, Startup Tests
and other items which must be completed
by the Indicated Operational Mode

Date of Issuance: July 16, 1990

MARKUP OF CURRENT ANO-2 LICENSE

(FOR INFORMATION ONLY)

2.C.(4) (Number has never been used.)

(5) EOI shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following.

1. Provisions establishing preventative maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(6) EOI shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

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(10) Cycle 14 Risk-Informed Operation

For Cycle 14 only, Entergy Operations shall be permitted to operate the reactor based on a risk-informed demonstration that predicted steam generator tube integrity, with consideration of eggcrate axial flaws, is adequate to meet Regulatory Guide 1.174 numerical acceptance criteria. In accordance with Principle 5 in Regulatory Guide 1.174 concerning monitoring operational experience to ensure that performance is consistent with risk predictions, if Entergy Operations plugs or repairs steam generator tubes during Cycle 14, then the steam generators shall be reinspected to the extent necessary to verify that they have been returned to a condition consistent with the risk assessment.

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Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Industrial Security Plan," with revisions submitted through August 4, 1995. The Industrial Security Plan also includes the requirements for guard training and qualification in Appendix A of the safeguards contingency events in Chapter 7. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

FORM TITLE: CALCULATION REVISION SHEET	FORM NO. 5010.015-ATT-2	REV. 1 PC-2

This Document contains 1 Page.

Check if Additional Revisions: ☐

Calc. No.:	99-E-0019-01	Rev. No.:	1

Calc. Title: ANO-2 Cycle-14 Steam Generator Tube Rupture Risk Assessment	Unit: 2	Category: Non-Q
	Config. Checklist (per 5010.004) completed? (Y or N)	Y
	Document Comment/Resolution Form completed? (Y or N)	~~N~~ Y 3/14/00

Component No(s) added/deleted:
None

Special changes to the IDEAS/DCIMS data base:
None

Abstract (Complete only if Abstract is changed by this revision):

This calculation documents a Steam Generator Tube Rupture (SGTR) risk assessment performed in support of an ANO-2 Steam Generator (SG) Operational Assessment for the second half of ANO-2 Cycle-14 in response to Condition Report CR-ANO-2-1999-0727. The risk assessment quantified the risk benefits associated with a proposed SG Inspection/Repair outage (called 2P00) during the second half of Cycle-14. These assessments estimate the risk associated with spontaneous Steam Generator Tube Ruptures (SGTRs), Pressure-Induced SG Tube Ruptures (PI-SGTRs), and Temperature-Induced SGTRs (TI-SGTRs). These events were identified in NUREG-1570 as the primary risk contributors associated with SG tube ruptures. The Core Damage Frequency (CDF), change in CDF (ΔCDF), Large Early Release Frequency (LERF), and the change in LERF (ΔLERF) over the second half of ANO-2 Cycle-14 were calculated. These values were compared with NRC risk acceptance guidelines for these parameters provided in NRC Regulatory Guide 1.174 to determine the most appropriate ANO-2 SG operating strategy. The risk associated with operating ANO-2 from 2P99 to 2R14 without a SG inspection/repair outage is considered small and therefore acceptable from a risk perspective.

Pages Revised and/or Added:
All: Pages 1 through 41, A-1 through A-4, B-1 through B-3, C-1 through C-4, D-1 through D-12, E-1 through E-9, F-1 through F-3, G-1 through G-8, H-1 through H-5

Purpose of Revision:
Clarification and correction of typographical errors

Initiating Documents	Resulting Documents	Key Design Input Documents
None	None	99-E-0019-01, Rev. 0

Verification Method: Design Review ☒ Alternate Calculation ☐ Qualification Testing ☐

Amends Calc(s):	none
Supersedes Calc(s):	none
Computer Software Used:	CAFTA, WORD, EXCEL

By:	M. Lloyd	ML	3/9/00	Rvw'd:	n/a		
Chk'd	D. W. Fouts	DWF	3/9/00	Apv'd:	D. W. Fouts	DWF	3/9/00
	(Print Name)	(Initials	(Date)		(Print Name)	(Initials	(Date)

1.0 PURPOSE

This calculation documents a Steam Generator Tube Rupture (SGTR) risk assessment performed in support of an ANO-2 Steam Generator (SG) Operational Assessment for the second half of ANO-2 Cycle-14 in response to Condition Report CR-ANO-2-1999-0727. The risk assessment quantified the risk benefits associated with a proposed SG Inspection/Repair outage (called 2P00) during the second half of Cycle-14. These assessments estimate the risk associated with spontaneous Steam Generator Tube Ruptures (SGTRs), Pressure-Induced SG Tube Ruptures (PI-SGTRs), and Temperature-Induced SGTRs (TI-SGTRs). These events were identified in NUREG-1570 [Ref. 1] as the primary risk contributors associated with SG tube ruptures. The Core Damage Frequency (CDF), change in CDF (ΔCDF), Large Early Release Frequency (LERF), and the change in LERF (ΔLERF) over the second half of ANO-2 Cycle-14 were calculated. These values were compared with NRC risk acceptance guidelines for these parameters provided in NRC Regulatory Guide 1.174 [Ref. 2] to determine the most appropriate ANO-2 SG operating strategy.

2.0 REFERENCES

1. NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Ruptures", USNRC/NRR, March 1998.
2. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis", July 1998.
3. Letter from J. J. Fisicaro (ANO) to USNRC, 2CAN089201, "Arkansas Nuclear One - Unit 2, Individual Plant Examination for Severe Accident Vulnerabilities – Generic Letter 88-20", 8/28/92.
4. 93-E-0079-05, Rev. 0, "ANO-2 SG Inspection Interval Risk Analysis Based On Inspection Data Through 2R10".
5. E. L. Fuller, et. al., "Steam Generator Tube Integrity Risk Assessment Methodology", EPRI TR-107623, Vol. 1, April 1999.
6. NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity", September 1988.
7. 95-R-1010-02, Rev. 0, "ANO-2 Probabilistic Safety Analysis (Rev. 1) Summary".
8. E. L. Fuller, et. al., "Assessment of Risk from Thermal Challenge to Steam Generator Tubes during Hypothetical Severe Accidents: Diablo Canyon as an Example Plant", EPRI Project S550-18, Draft Report, September, 1997.
9. 97-E-0019-02, Rev. 0, "ANO-2 MAAP and PROBFAIL Calculations".
10. 97-E-0036-01, Rev. 0, "ANO-2 EOOS Master PSA Model Development".
11. 89-E-0048-20, Rev. 2, "ANO-2 PRA Initiating Events and Accident Sequence Analysis Work Package".
12. Analysis Report CEO-92/00170, "ANO-2 MAAP Analyses to Support the CE Owners Group Task for Severe Accident Management", 3/17/92.
13. 89-E-0048-26, Rev. 2, "ANO-2 ATWS Scoping Report".
14. 89-E-0048-22, Rev. 1, "ANO-2 Human Reliability Analysis (HRA) Work Package".
15. 97-E-0036-02, Rev. 0, "ANO-2 EOOS Monitor Implementation".

3.0 ASSUMPTIONS

1. The risk assessment documented in this calculation addresses the risk impact of axial SG defects (i.e., "eggcrate" defects); it does not address the risk impact of other defect types.
2. SG tube rupture events and Interfacing System LOCA events are conservatively assumed to be Larger Early Release events.
3. The frequency of a spontaneous SGTR is assumed to remain constant as a function of operating time, no change in CDF or LERF due to a spontaneous SGTR is expected.
4. It is conservatively assumed that the FLB and SLB both result in a pressure difference across the SG tubes of 2500 psid.
5. It is assumed that the ATWS event produces a range in RCS pressures depending on the cycle burnup. The maximum pressure of the RCS due to an ATWS event is assumed to be 3700 psia.
6. The ATWS analysis assumes that the SG secondary pressure is 528 psia for 85% of ATWS cases and that they are fully depressurized to atmospheric pressure (via a stuck-open MSSV) for the remaining 15% of ATWS cases.
7. Best estimate Steam Generator (SG) tube burst probability values were utilized in this calculation. These values were based on data collected during ANO-2 SG Inspection/Repair Outage 2P99. The values were generated as a function of the pressure difference between the primary and secondary regions and as a function of burnup for the most limiting Steam Generator during ANO-2 Cycle 14.
8. The Middle of Period with a SG inspection/repair outage (MOP-WR) SG pressure fragility value is assumed to be the same as that at the Beginning of Period (BOP) value at 2P99; and, the End of Period (2R14) with Repairs (EOP-WR) value is based on interpolating or extrapolating the SG fragility from the MOP to the EOP burnup using the BOP and Middle of Period with no Repair (MOP-NR) values.
9. The Beginning of Period (BOP) is assumed to start immediately after 2P99 on 11/15/99. The proposed Middle of Period (MOP) SG inspection/repair outage 2P00 is assumed to begin on 5/15/00. And, the End of Period (EOP) is assumed to occur at the end of Cycle-14 (2R14) on 9/15/00.
10. Temperature Induced SGTRs are assumed to require dry SG conditions.
11. It was assumed that the Internal Events CDF "split fractions" on RCS pressure and SG inventory condition apply to both the Internal Flooding and External Events accidents.
12. If ADV fails, it was assumed that there are 85 MSSV open demands in an accident involving a High RCS pressure and loss of all feedwater. It was arbitrarily assumed that only 25 MSSV demands would occur prior to the use of the ADVs.
13. Event SG_BOTTLE represents the failure to completely "bottle" both SGs, i.e., isolate all but the ADV path, prior to the SGs dry. It assumes that the probability that the support systems required to perform the valve isolations are accounted for in the ADV support system assessment. This assumption takes no credit for manual actions to close the subject isolation valves.
14. Event *RCS_INTEG* accounts for the loss of RCS integrity due to stuck open primary Safety Relief Valve (SRV) or an intentional RCS depressurization by an operator. The success of this event reduces the RCS pressure, reduces the transport of heat to the SGs, and reduces the

pressure difference across the SG tubes. Although the event is likely to fully depressurize the RCS, it was conservatively assumed that the event depressurizes the RCS to a "Medium" pressure, i.e. to about 1400 psia.

15. For event PSRVSO, the loss of RCS integrity due to a stuck open (SO) Primary Safety Relief Valve (PSRV), a probability of 0.14 was assumed for a SO PSRV prior to core uncovery and a probability of 0.5 was assumed after core damage.

4.0 ANALYSIS

4.1 Background

NUREG-1570 [Ref. 1] identified three primary contributors to Steam Generator Tube Ruptures (SGTRs):

(1) spontaneous Steam Generator Tube Ruptures (Sp-SGTRs) occurring during normal operation,
(2) pressure transient-induced or pressure induced SGTRs (PI-SGTRs) resulting from primary-to-secondary differential pressure conditions caused by a design-basis transient or accident, and
(3) core damage-induced or temperature induced SGTRs (TI-SGTRs) resulting from a core damage condition

The spontaneous SGTR risk assessment is based on the ANO-2 Rev. 1 PSA results provided in Ref. 3; the pressure-induced SGTR risk assessment is based on the EPRI SG degradation-specific management program methodology as applied to ANO-2 and documented in Ref. 4; and, the temperature-induced SGTR risk assessment is consistent with the EPRI SG tube integrity risk assessment methodology [Ref. 5] and use of the ANO-2 Rev. 1 PSA results.

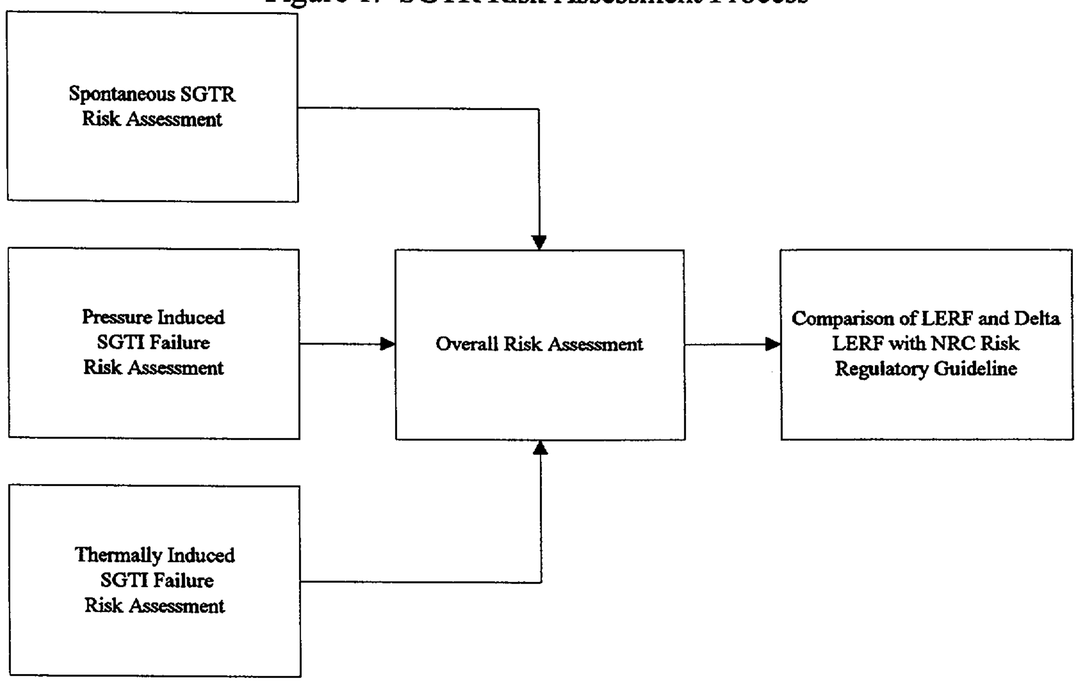
Since the SG materiel condition affects the likelihood of a SGTR and since this condition is expected to deteriorate between SG inspection/repair outages, the ANO-2 TI-SGTR Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) were calculated at several points in burnup during the second half of ANO-2 Cycle-14.

(1) Immediately after the 2P99 SG inspection/repair campaign (called the "Beginning of Period (BOP)"),
(2) Immediately before a proposed SG inspection/repair campaign. 2P00 (called "Middle of Period, with No 2P00 SG Repair (MOP-NR)"),
(3) Immediately before the scheduled Refueling outage at the End of Cycle 14 (called the "End of Period, with No 2P00 Repair (EOP-NR)"),
(4) Immediately after the proposed 2P00 SG inspection/repair campaign (called "Middle of Period With 2P00 SG Repair (MOP-WR)"), and
(5) Immediately before the scheduled 2R14 Refueling outage at the End of Cycle 14 (called the "End of Period, With 2P00 SG Repair (EOP-WR)").

The CDF and LERF were assessed using best-estimate ANO-2 plant-specific SG tube defect data associated at each burnup condition.

The Core Damage Frequency (CDF), change in CDF (ΔCDF), Large Early Release Frequency (LERF), and the change in LERF (ΔLERF) for the BOP-to-MOP and for the MOP-to-EOP operating periods were calculated. These values were compared with NRC risk acceptance guidelines for these parameters provided in NRC Regulatory Guide 1.174 [Ref. 2] to determine the acceptability of the SG inspection strategy. The approaches used to assess the risk and change in risk associated with spontaneous SGTRs, PI-SGTRs, and TI-SGTRs are presented in Sections 4.2, 4.3, and 4.4, respectively. Figure 1, below, outlines the overall risk assessment process.

Figure 1. SGTR Risk Assessment Process



During the ANO-2 2P99 SG Inspection/Repair Outage, all axial SG defects were inspected and all confirmed defects were plugged. The risk assessment documented in this calculation addresses the risk impact of axial SG defects (i.e., "eggcrate" defects); it does not address the risk impact of other defect types.

4.2 Spontaneous SGTR Risk Analysis

Spontaneous SGTRs are those occurring during power operation which are not due to significant changes in the primary-to-secondary pressure differential. Per Ref. 1,

> "the risk from spontaneous and pressure transient-induced SGTRs was previously assessed by the staff in NUREG-0844 [Ref. 6]. More recent assessments have shown that if measures are implemented to maintain tube integrity consistent with current requirements, no significant change is expected in the risk from these contributors (Ellison, 1996)."

These "measures" consist of periodic ANO-2 SG inspection and repair campaigns. For the current ANO-2 SGs, these SG inspection/repair campaigns are performed twice per operating cycle. Consistent with these measures, SG defects are either repaired or removed from service prior to the point that they are large enough to result in a significant increase in the spontaneous SGTR frequency. The ANO-2 spontaneous SGTR frequency reported in the ANO-2 Individual Plant Examination (IPE) [Ref. 3] and used in the ANO-2 PSA Model, Rev-1 [Ref. 7] is 9.77E-3/rx-yr. The CDF associated with the spontaneous SGTR reported in Table 6 of Ref. 7 is 1.398E-07; since the spontaneous SGTR is assumed to be a LERF, the LERF of this event has this same value. Since no change in the spontaneous SGTR frequency is expected, the frequency of a spontaneous SGTR is assumed to remain constant as a function of operating time. Thus, no change in CDF or LERF due to a spontaneous SGTR is expected.

4.3 Pressure-Induced SGTR Risk Analysis

Pressure Induced SGTRs (PI-SGTRs) are SGTRs which occur during power operation as a result of a significant increase in the pressure difference between the primary and secondary sides of the SGs. Per Section 2.1 of Ref. 1, pressure induced SGTR challenges could result from either a secondary side depressurization or primary system over-pressurization. The former include the Feedwater Line Break (FLB), the Steam Line Break (SLB), and transients with a stuck open secondary relief valve; the latter includes the ATWS event. The FLB and SLB pressure induced SGTR risk assessments are presented in Section 4.3.1. The ATWS pressure induced SGTR risk assessments are presented Section 4.3.2. Transients with a stuck open secondary relief valve are assumed to be dominated by the risk associated with the temperature induced SGTR risk. The latter is presented in Section 4.4.

4.3.1 FLB/SLB-Induced SGTR Risk Analysis

The Ref. 4 methodology and best estimate probability of SG tube burst values reported in Attachment B were applied to assess the FLB and SLB PI-SGTR CDFs and LERFs during the second half of ANO-2 Cycle-14. Consistent with the methodology used in Ref. 4, the FLB and SLB were treated together. Also, consistent with the Ref. 4, it is assumed that the FLB and SLB both result in a pressure difference across the SG tubes of 2500 psid. This pressure difference is conservatively high, since it can occur only if the SG is depressurized to atmospheric pressure and if the RCS is pressurized to the Pressurizer relief valve setpoint of 2500 psig. The best-estimate BOP, MOP-NR, and EOP-NR SG tube burst failure probability values reported in Ref. 9 and repeated in Table B-1 of Attachment B (i.e., 0.000414, 0.00097, 0.002080, respectively) were substituted for the 0.368801 value applied in Table E-3 of Ref. 4 to generate the FLB/SLB CDF estimates for BOP, MOP-NR, and EOP-NR. These PI-SGTR probabilities were used to estimate values for MOP-WR and for EOP-WR. The MOP-WR value is assumed to be the same as the BOP value; and, the EOP-WR value is based on interpolating or extrapolating the PI-SGTR probability from the MOP to the EOP burnup using the BOP and MOP-NR PI-SGTR values. Note all of the new SG tube burst probabilities are significantly lower than that assumed for

BOC11 in Ref. 4. The lower estimates for the PI-SGTR in Table B-1 are based on actual plant data collected during 2P99; whereas, the Ref. 4 value was a conservative estimate using information available at the time.

Using the EXCEL spreadsheet FLBSLB-2P99.xls (a modified version of the Ref. 4 spreadsheet SGTRCDF-SG.XLS), the FLB/SLB induced SGTR instantaneous CDFs were calculated at BOP, MOP-NR, EOP-NR, MOP-WR and EOP-WR. These results are reported in Table 1. The interval average CDFs between these points are reported in Table 2. Spreadsheet FLBSLB-2P99.xls is provided in Attachment A.

Table 1. FLB/SLB-induced SGTR Instantaneous CDFs

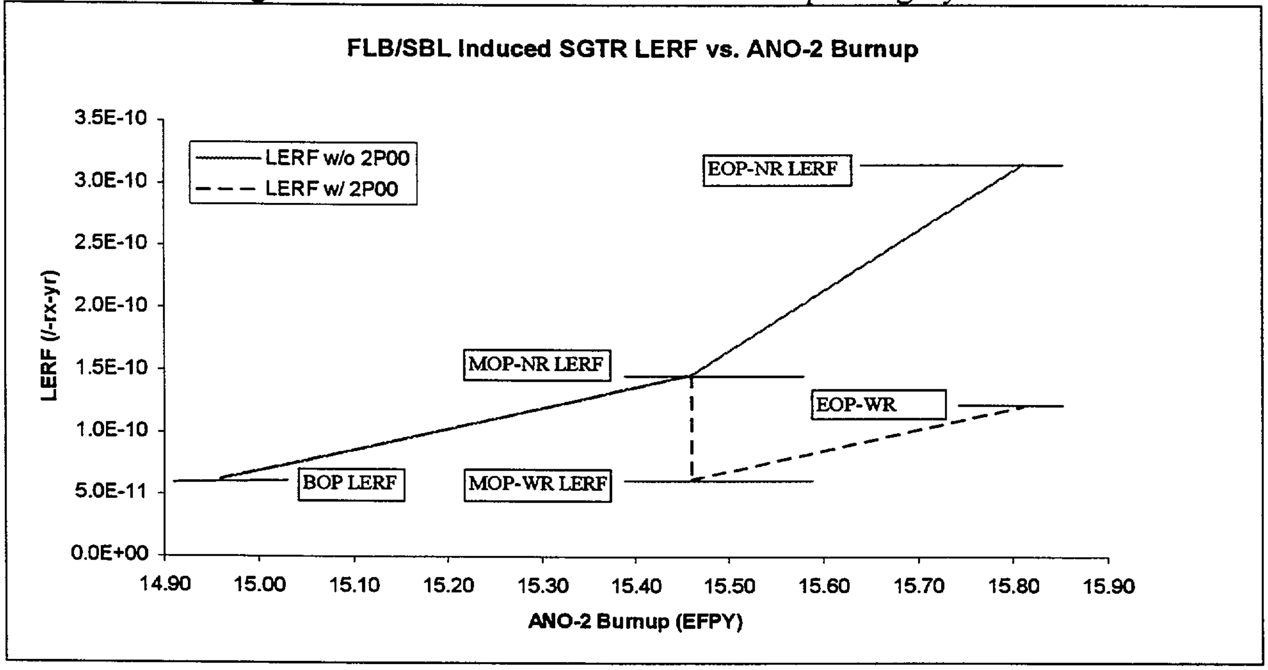
ANO-2 Burnup	Instantaneous CDF FLB/SLB Induced SGTR (/rx-yr)
BOP	6.252E-11
MOP-NR	1.466E-10
EOP-NR	3.153E-10
MOP-WR	6.252E-11
EOP-WR	1.214E-10

Table 2. FLB/SLB-induced SGTR Average CDFs

ANO-2 Burnup Interval	Interval Average CDF FLB/SLB Induced SGTR (/rx-yr)
BOP to MOP-NR	1.046E-10
MOP-NR to EOP-NR	2.310E-10
MOP-WR to EOP-WR	9.196E-11

Consistent with the classification of a spontaneous SGTR, the FLB/SLB-induced SGTR core damage event is considered a Large Early Release. Thus, the FLB/SLB-induced Large Early Release Frequency (LERF) values are equal to their corresponding CDF values. Thus, the FLB/SLB-induced CDF values reported in Tables 1 and 2 are LERFs. Figure 2 shows the behavior of the FLB/SLB LERF over the second half of ANO-2 Cycle-14.

Figure 2. ANO-2 FLB/SLB LERF vs. Burnup during Cycle-14



4.3.2 ATWS-Induced SGTR Risk Analysis

The Ref. 4 methodology and best estimate probability of SG tube burst values reported in Ref. 9 and repeated in Table B-1 of Attachment B were applied to assess the ATWS PI-SGTR CDFs and LERFs during the second half of ANO-2 Cycle-14. Consistent with the Ref. 4, it is assumed that the ATWS event produces an RCS pressure which depends on the cycle burnup. The burnup dependent ATWS RCS pressure results calculated in Ref. 4 were used in this analysis. Consistent with Ref. 4, the maximum pressure of the RCS due to an ATWS event is assumed to be 3700 psia. Consistent with the Ref. 4 analysis, this analysis assumes that the SG secondary pressure is 528 psia for 85% of ATWS cases and that they are fully depressurized to atmospheric pressure (via a stuck-open MSSV) for the remaining 15% of ATWS cases. The pressure-dependent SG tube burst failure probability values at BOP, MOP-NR, and EOP-NR reported in Table B-1 of Attachment B were substituted for the values used in Table F-4 of Ref. 4 to generate the ATWS CDF and LERF estimates at BOP, MOP-NR, and EOP-NR. In addition, ATWS CDF and LERF estimates were generated at MOP-WR and EOP-WR. These PI-SGTR probabilities were generated using the BOP-NR and MOP-NR values. As was done in the FLB/SLB analysis, the MOP-WR PI-SGTR value was assumed to be the same as the BOP value; and, the EOP-WR value is based on interpolating or extrapolating the PI-SGTR probability from the MOP to the EOP burnup using the BOP and MOP-NR PI-SGTR values.

Using the EXCEL spreadsheet ATWS-2P00.xls (a modified version of the Ref. 4 spreadsheet ATWS1.XLS), the ATWS induced SGTR instantaneous CDFs were calculated at BOP, MOP-NR, EOP-NR, MOP-WR, and EOP-WR. These results are reported in Table 3. The interval average CDFs between these points are reported in Table 4. Note that the interval average CDFs have been averaged over many small burnup intervals rather than on a single interval; thus, the

CDF values are not half of the difference between their beginning and ending CDFs. Spreadsheet ATWS-2P99.xls is provided in Attachment A.

Table 3. ATWS-induced SGTR Instantaneous CDFs

ANO-2 Burnup	Instantaneous CDF ATWS Induced SGTR (/rx-yr)
BOP	6.635E-09
MOP-NR	5.935E-09
EOP-NR	7.569E-09
MOP-WR	2.937E-09
EOP-WR	2.956E-09

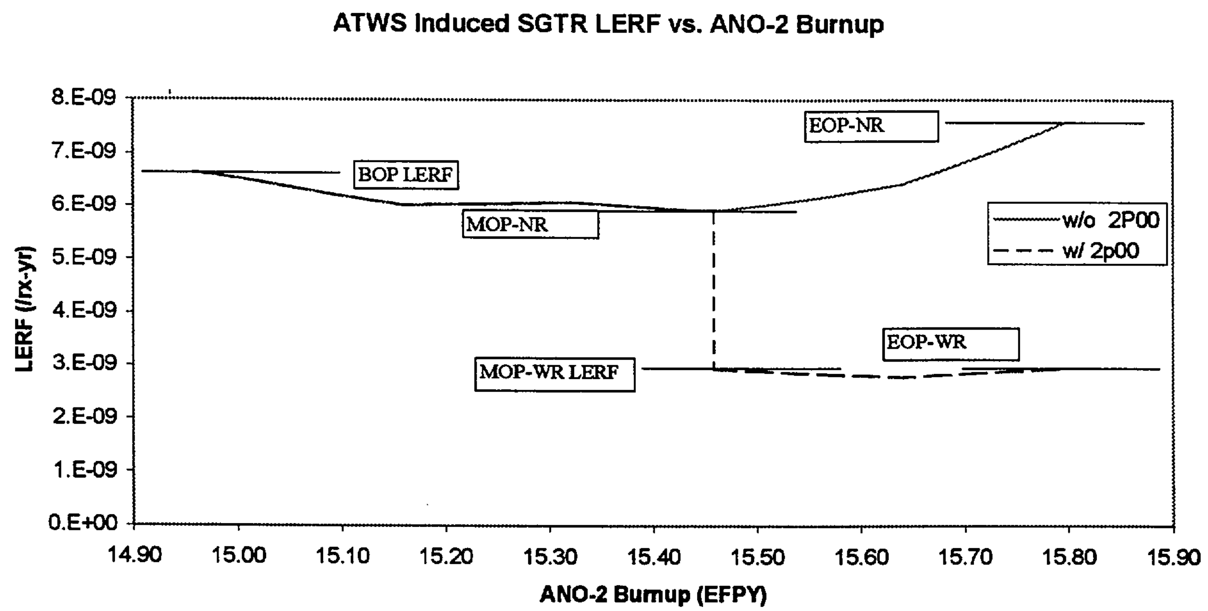
Table 4. ATWS-induced SGTR Average CDFs

ANO-2 Burnup Interval	Interval Average CDF ATWS Induced SGTR (/rx-yr)
BOP to MOP-NR	6.149E-09
MOP-NR to EOP-NR	6.534E-09
MOP-WR to EOP-WR	2.864E-09

Consistent with the classification of a spontaneous SGTR, the ATWS-induced SGTR core damage event is considered a Large Early Release. Thus, the ATWS-induced Large Early Release Frequency (LERF) values are equal to their corresponding CDF values. Thus, the ATWS-induced CDF values reported in Tables 3 and 4 are LERFs.

Figure 3 shows the behavior of the ATWS Induced SGTR LERF over the second half of ANO-2 Cycle-14.

Figure 3. ANO-2 ATWS LERF vs. Burnup during Cycle-14



4.3.3 Combined FLB/SLB and ATWS-Induced SGTR Risk Impact

The total PI-SGTR risk impact is that due to FLBs, SLBs, and ATWS-Induced SGTR events combined. Based on the results presented in Sections 4.3.1 and 4.3.2, the change in CDF due to these events is summarized in Tables 5 and 6.

Table 5. FLB/SLB-induced and ATWS-induced SGTR Instantaneous CDFs

ANO-2 Burnup	**Instantaneous CDF FLB/SLB + ATWS Induced SGTR (/rx-yr)**
BOP	6.697E-09
MOP-NR	6.082E-09
EOP-NR	7.884E-09
MOP-WR	2.999E-09
EOP-WR	3.077E-09

Table 6. FLB/SLB-induced and ATWS-induced SGTR Average CDFs

ANO-2 Burnup Interval	**Interval Average CDF FLB/SLB + ATWS Induced SGTR (/rx-yr)**
BOP to MOP-NR	6.254E-09
MOP-NR to EOP-NR	6.765E-09
MOP-WR to EOP-WR	2.956E-09

As noted above, these core damage events are considered Large Early Releases. Thus, the PI-SGTR CDF values reported in Tables 5 and 6 are LERF values.

4.4 Temperature-Induced SGTR Risk Analysis

Temperature Induced SGTRs (TI-SGTRs) are SG tube ruptures caused by the heating of SG tubes by hot gases released from a damaged core. The likelihood of a TI-SGTR depends on both plant design and on accident conditions. Since severe accident conditions must exist for a TI-SGTR event, the TI-SGTR does not increase core damage frequency. Rather, TI-SGTRs affect only the likelihood and magnitude of a fission product release to the environment during a severe accident by creating a release path from the damaged core and through the RCS/SG boundary. The design of the RCS hot legs, the SGs, and the Reactor Coolant Pump (RCP) seals have been identified [Ref. 1] to affect the probability of a TI-SGTR during a severe accident. The severe accident conditions which pose the greatest potential to a TI-SGTR are those in which the RCS pressure is high, the SG secondary side is dry, the SG secondary side pressure is low, and when the RCP loop seal is clear.

The ANO-2 TI SGTR risk assessment was performed using the EPRI SG tube integrity risk assessment methodology [Ref. 5]. The process included use of the existing ANO-2 Rev. 1 PSA results, plant-specific Emergency Operating Procedures (EOPs), and Severe Accident Management Guidelines (SAMGs). Consistent with this methodology, the frequency of a TI-SGTR at ANO-2 was assessed by identifying the ANO-2 core damage accidents that are vulnerable to TI-SGTRs, assessing their frequency, and evaluating the probability that a TI-SGTR will occur at ANO-2 given these severe accident conditions.

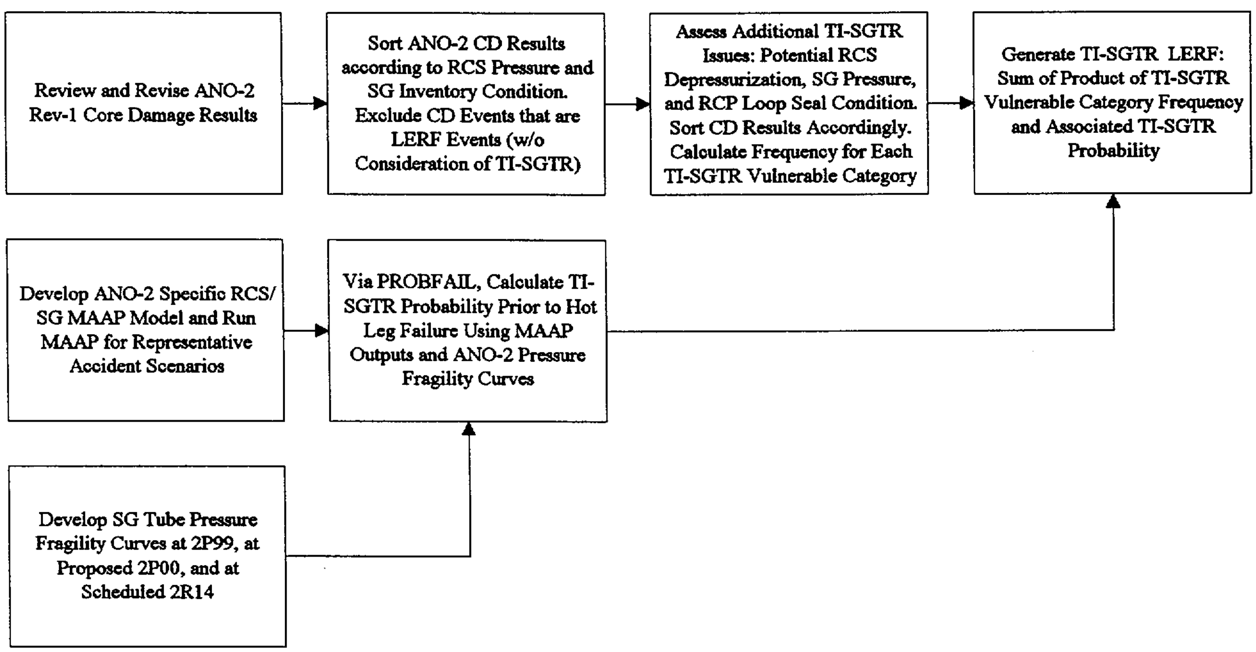
The first step of the process involved a review of the existing ANO-2 Rev-1 core damage results. This review indicated that the current results overestimated the frequency of core damage, especially the frequency of core damage involving a high RCS pressure and a dry SG condition. The results were revised to account for additional operator recoveries. Then, the ANO-2 core damage results were sorted according to RCS pressure and SG inventory condition. Core damage scenarios which are already LERF events for reasons other than their vulnerability to a TI-SGTR were excluded from further TI-SGTR assessment.

The current ANO-2 core damage results do not describe the status of the SG pressure, the potential for the RCS to depressurize, and the potential loss of the RCP loop seal during the core damage accident progression, since TI-SGTR issues were not considered in the ANO-2 Level-1 (CDF) or Level-2 (LERF) risk assessments. Since these parameters have been identified to affect the probability of a TI-SGTR, a separate assessment of their status was performed. Using the results of this assessment, the ANO-2 core damage scenarios involving high RCS pressure and a dry SG were further sorted according to their SG pressure condition, and according to whether they are likely to involve an RCS depressurization event or the loss of an RCP loop seal. This assessment produced a frequency for each TI-SGTR vulnerability group defined by RCS pressure/SG inventory/SG pressure/RCP seal loop condition.

In the next step of the ANO-2 TI-SGTR risk assessment, the likelihood of a TI-SGTR was assessed for each of the vulnerability groups utilizing ANO-2 plant-specific MAAP analyses with subsequent PROBFAIL calculations. The PROBFAIL code uses the MAAP accident simulation results and the actual SG degradation conditions to estimate the probability that a SGTR event occurs prior to a RCS hot leg failure. The development and use of a plant-specific MAAP and PROBFAIL models are outlined in Ref. 5.

For each TI-SGTR vulnerability group, an estimate for the TI-SGTR frequency was generated by combining the TI-SGTR group frequency with the PROBFAIL probability of a TI-SGTR results. The total TI-SGTR frequency was calculated by summing the TI-SGTR frequencies of the TI-SGTR vulnerability groups. A TI-SGTR event was conservatively assumed to be a Large Early Release event. Thus, the generated TI-SGTR frequency is assumed to be a contributor to the ANO-2 Large Early Release Frequency (LERF). An overview of the ANO-2 TI-SGTR risk assessment process is depicted in Figure 4, below.

Figure 4. TI-SGTR Risk Assessment Process



Section 4.4.1 documents the evaluation of the TI-SGTR probability for a spectrum of RCS pressure, SG pressure, and RCP Loop Seal conditions. Section 4.4.2 documents the sorting of ANO-2 core damage accidents vulnerable to TI-SGTR. Specifically, the section documents the sorting of the ANO-2 CDF accidents according to RCS pressure and SG inventory condition and estimates the frequency of each these accident groups. In addition, this section documents the assessment of the probability that a core damage scenario will involve a given SG pressure, a RCS depressurization event, and an RCP seal LOCA condition. The results of Sections 4.4.1 and 4.4.2 are combined in Section 4.4.3 to provide a measure of the TI-SGTR effect on the ANO-2 Large Early Release Frequency (LERF).

4.4.1 Evaluation of TI-SGTR Probability for Various Plant Conditions

NUREG-1570 identified the Combustion-Engineering (CE) Nuclear Steam Supply System (NSSS) to be particularly susceptible to the TI-SGTR issue due to their relatively large hot leg diameters, relatively thick hot leg piping, U-tube SGs, the relatively shallow primary side inlet SG plenums. The large hot legs, shallow primary side SG inlet plenums, and U-tube design are expected to allow the establishment of natural convection flow between the melting core and relatively cool SGs. If RCP loop seals are intact, a counter-current natural convection flow is expected in the hot legs between the melting core and relatively cool SGs. The hot gases exit the reactor vessel, flow into and along the top of the hot legs and eventually into their associated SG. The gases then plume into the SG tubes just above the junction of the hot leg and the SG; the gases chimney through these SG tubes into the cold leg SG plenum, return along other SG tubes, and return to the reactor vessel along the bottom of the hot leg. If one or more RCP seal loops is not intact, the hot gases exit the reactor vessel, flow into the hot leg, plume into the SG tubes just above the junction of the hot leg, pass these tubes into the cold leg SG plenum, into the cold leg without a RCP loop seal associated with the SG, and return to the reactor vessel via the cold leg. Since the former flow path involves hot leg counter-current flow and the latter one-directional flow, the latter is expected to produce greater flow rates of hot gases from the core than the former.

The higher the RCS pressure, the more likely and efficient the natural convention process is expected to be. This process is likely to be weak or non-existent for larger LOCAs which depressurize the RCS. In addition, the stress on the SG tubes increases as the pressure difference across the SG tubes increases; thus, the higher the RCS pressure and the lower the SG secondary pressure, the more likely the TI-SGTR. A dry SG condition is required for high SG tube temperatures and for creep failure of the tubes. It should be noted that for all but the larger LOCA events, dry SGs are likely to occur prior to significant core damage. The presence of significant SG tube degradation prior to a severe accident is also expected to increase the probability of a TI-SGTR.

Consistent with the EPRI SG tube integrity risk assessment methodology, the best-estimate probability of burst estimates generated as a function of RCS/SG differential pressure for several burnup points in the second half of ANO-2 Cycle-14 were used as input to the PROBFAIL calculations. As noted in Section 4.1, these burnup points are:

(1) Immediately after the 2P99 SG inspection/repair campaign (called the "Beginning of Period (BOP)"),
(2) Immediately before a proposed planned SG inspection/repair campaign. 2P00 (called "Middle of Period, with No 2P00 SG Repair (MOP-NR)"),
(3) Immediately before the scheduled Refueling outage at the End of Cycle 14 (called the "End of Period, with No 2P00 Repair (EOP-NR)"),
(4) Immediately after the proposed 2P00 SG inspection/repair campaign (called "Middle of Period With 2P00 SG Repair (MOP-WR)"), and
(5) Immediately before the scheduled 2R14 Refueling outage at the End of Cycle 14 (called the "End of Period, With 2P00 SG Repair (EOP-WR)").

As noted in Section 4.3, the first three of the probability estimates are documented in Attachment B of Ref. 9 and repeated in Attachment B of this calculation. As was done in the FLB/SLB analysis, the MOP-WR PI-SGTR value was assumed to be the same as the BOP value; and, the EOP-WR value is based on interpolating or extrapolating the burst probability from the MOP to the EOP burnup using the BOP and MOP-NR burst probability values.

Consistent with the EPRI methodology, an ANO-2 plant-specific MAAP 4.0.3 model was developed for the TI-SGTR analysis. This model was developed by combining the RCS/SG part of the existing ANO-2 MAAP 3 model with the containment portion of the Zion MAAP 4.0.3 model. The hybrid ANO-2 MAAP 4 model was completed by including additional ANO-2 plant-specific RCS/SG information into to the MAAP 4 parameter file. This process is briefly described in Appendix B of the EPRI Diablo Canyon TI-SGTR assessment, Ref. 8. The development of the hybrid ANO-2 MAAP 4 model is documented in Ref. 9.

Reference 9 documents the ANO-2 MAAP calculations performed in support of the ANO-2 TI-SGTR risk assessment. These calculations consist of MAAP simulations for a spectrum of postulated severe accidents for ANO-2 and subsequent PROBFAIL calculations which use the MAAP output to calculate the probability of a TI-SGTR prior to hot leg failure. PROBFAIL is described in Ref. 5. Briefly, this code uses MAAP output (including the RCS gas temperature in the RCS hot leg and the RCS gas temperature in the SG tubes) during the accident scenario. The code uses the MAAP-generated gas temperatures to calculate a time-dependent temperature profile of the hot leg and then uses this profile to calculate the hot leg failure pressure. It also calculates the probability of a TI-SGTR using the temperature-adjusted best-estimate ANO-2 plant-specific SG pressure "fragility" distribution supplied in Attachment B. It then compares these failure pressures to estimate the probability of a SG tube failure prior to a hot leg failure.

The results of the MAAP and PROBFAIL TI-SGTR calculations for second half of ANO-2 Cycle-14 are provided in Table 5.2.2 of Reference 9. These results are repeated in Table 7, below. They consist of the ANO-2 TI-SGTR probability for a spectrum of postulated severe accidents for the BOP, MOP-NR, and EOP-NR cases. Since MAAP calculations were not performed for all cases of interest (namely Cases 10 through 17), available Reference 9 cases were used to estimate other results as described in the footnote to the table.

Figure 5 depicts graphically the effect of RCS pressure, SG pressure, and RCP loop seal condition on the probability of a TI-SGTR as a function of burnup in the second half of ANO-2 Cycle14. The solid lines show the increase in TI-SGTR probability vs. burnup without a SG inspection/repair outage between 2P99 and 2R14. The dotted lines branch from the solid lines at the proposed MOP SG inspection/repair outage 2P00 (assumed to occur 5/15/00).

The MAAP/PROBFAIL results indicate that core damage with High RCS pressure/Dry SG/Low SG pressure (H/D/L) conditions is very likely to result in a TI-SGTR. Note that the "No SG Repair (NR)" TI-SGTR probabilities increase monotonically between 2P99 and 2R14, whereas, the "With SG Repair (WR)" TI-SGTR probabilities have a saw-toothed behavior with respect to Cycle-14 burnup. This behavior is consistent with a rise in TI-SGTR probability between SG inspections and a drop in the TI-SGTR probability after a SG inspection/repair campaign.

Note that Figure 5 follows Figure B-1 in that the probability of a SGTR is greater at EOP than at MOP (prior to inspection/repair). Figure 5 also shows the probability of a TI-SGTR is significantly reduced by a reduction in RCS pressure (from 2500 to 1400 psia) at the time of core damage. The probability of a TI-SGTR is predicted to decrease as the SG pressure increases; most of the reduction occurs between 14.7 and 550 psia.

The results also indicate that the effect of clearing an RCP loop seal varies, depending on the condition of the SG associated with the cleared loop. A cleared loop on a depressurized SG is expected to have a relatively high probability of TI-SGTR. On the other hand, a cleared loop on a fully pressurized SG is expected to have a relatively low probability of TI-SGTR. This result is believed to be due to the development of an relatively efficient natural convection cooling loop between the core and the SG on the open RCP loop; the flow through the open loop tends to rob heat transport through loops that remain intact. Thus, a cleared loop seal on a depressurized SG heats the affected SG tubes and hot leg relatively rapidly. The more rapid the heatup rate, the more likely SG tubes are likely to fail prior to the hot leg due to the relatively longer thermal lag of the hot leg piping. Whereas, although a fully pressurized SG on cleared loop heats quicker than that in an intact loop, relatively low pressure difference across the SG tubes allows the SG tubes to remain intact beyond the time of hot leg failure. In the asymmetric case of a fully pressurized SG on a cleared loop and a depressurized SG on an intact loop, the cleared loop appeared to actually reduce the probability of a SGTR; this is likely due to the cleared loop starving the intact loop of flow. An operating strategy using this result is not prudent at this time, however.

Table 7. ANO-2 Cycle-14 MAAP TI-SGTR Probability Results

Case	Case Description [1]	P_{RCS} (psia)	P_{SG1} (psia)	P_{SG2} (psia)	dP_{SG1} (psid)	dP_{SG2} (psid)	RCS Loop 1 Seal Condition	RCS Loop 2 Seal Condition	P_{SGTR} BOP[2] After Repair	P_{SGTR} MOP Before Repair	P_{SGTR} MOP After Repair	P_{SGTR} EOP Before Repair	P_{SGTR} Pristine
1[3]	H/D/Li/Li	2500	14.7	14.7	2485	2485	Intact	Intact	8.951E-01	9.373E-01	8.951E-01	9.616E-01	2.386E-02
2	H/D/Mi/Mi	2500	550	550	1950	1950	Intact	Intact	1.497E-01	2.091E-01	1.497E-01	2.801E-01	0.000E+00
3	H/D/Hi/Hi	2500	1100	1100	1400	1400	Intact	Intact	3.964E-03	7.088E-03	3.964E-03	1.229E-02	0.000E+00
4	M/D/Li/Li	1400	14.7	14.7	1385	1385	Intact	Intact	2.453E-01	3.069E-01	2.453E-01	3.733E-01	1.980E-04
5	H/D/Lc/Hi	2500	14.7	1100	2485	1400	Clear	Intact	9.358E-01	9.526E-01	9.358E-01	9.602E-01	9.200E-02
6	H/D/Li/Hc	2500	14.7	1100	2485	1400	Intact	Clear	1.202E-01	1.789E-01	1.202E-01	2.519E-01	0.000E+00
7	H/D/Hi/Li	2500	1100	14.7	1400	2485	Intact	Intact	6.762E-01	7.496E-01	6.762E-01	8.039E-01	1.200E-02
8	M/D/Hi/Li	1400	1100	14.7	300	1385	Intact	Intact	1.312E-01	1.675E-01	1.312E-01	2.084E-01	9.900E-05
9	H/D/Hi/Hc	2500	1100	1100	1400	1400	Intact	Clear	4.227E-02	6.648E-02	4.227E-02	1.000E-01	0.000E+00
10[4]	M/D/Mi/Mi	1400	550	550	850	850	Intact	Intact	4.103E-02	6.846E-02	4.103E-02	1.087E-01	0.000E+00
11	H/D/Li/Lc	2500	14.7	14.7	2485	2485	Intact	Clear	1.086E-03	2.321E-03	1.086E-03	4.772E-03	0.000E+00
12[5]	H/D/Lcmx	2500	14.7	14.7	2485	2485	clear	Intact	9.358E-01	9.526E-01	9.358E-01	9.602E-01	9.200E-02
13	H/D/Mcmx	2500	550	550	1950	1950	clear	Intact	1.565E-01	2.125E-01	1.565E-01	2.797E-01	0.000E+00
14	H/D/Hcmx	2500	1100	1100	1400	1400	clear	Intact	4.144E-03	7.204E-03	4.144E-03	1.227E-02	0.000E+00
15	M/D/Lcmx	1400	550	550	850	850	intact	Intact	2.564E-01	3.119E-01	2.564E-01	3.728E-01	7.635E-04
16	M/D/Mcmx	1400	550	550	850	850	intact	Intact	4.289E-02	6.958E-02	4.289E-02	1.086E-01	0.000E+00
17	M/D/Hcmx	1400	550	550	850	850	intact	Intact	1.135E-03	2.359E-03	1.135E-03	4.765E-03	0.000E+00

[1] Case descriptions are as follows: first entry describes RCS pressure condition: High (H) ~ 2500 psia, Medium (M) ~ 1400 psia, or Low (L) ~ 14.7 psia
second entry describes the SG inventory condition: Dry (D) or Wet (W)
third and fourth entries describe the SGA and SGB pressure conditions: High (H) ~ 1100 psia, Medium (M) ~ 550 psia, or Low (L) ~ 14.7 psia
the letter following the SG pressure entry denotes that at least one RCP loop seal associated with a SG is either intact (i) or cleared (c).

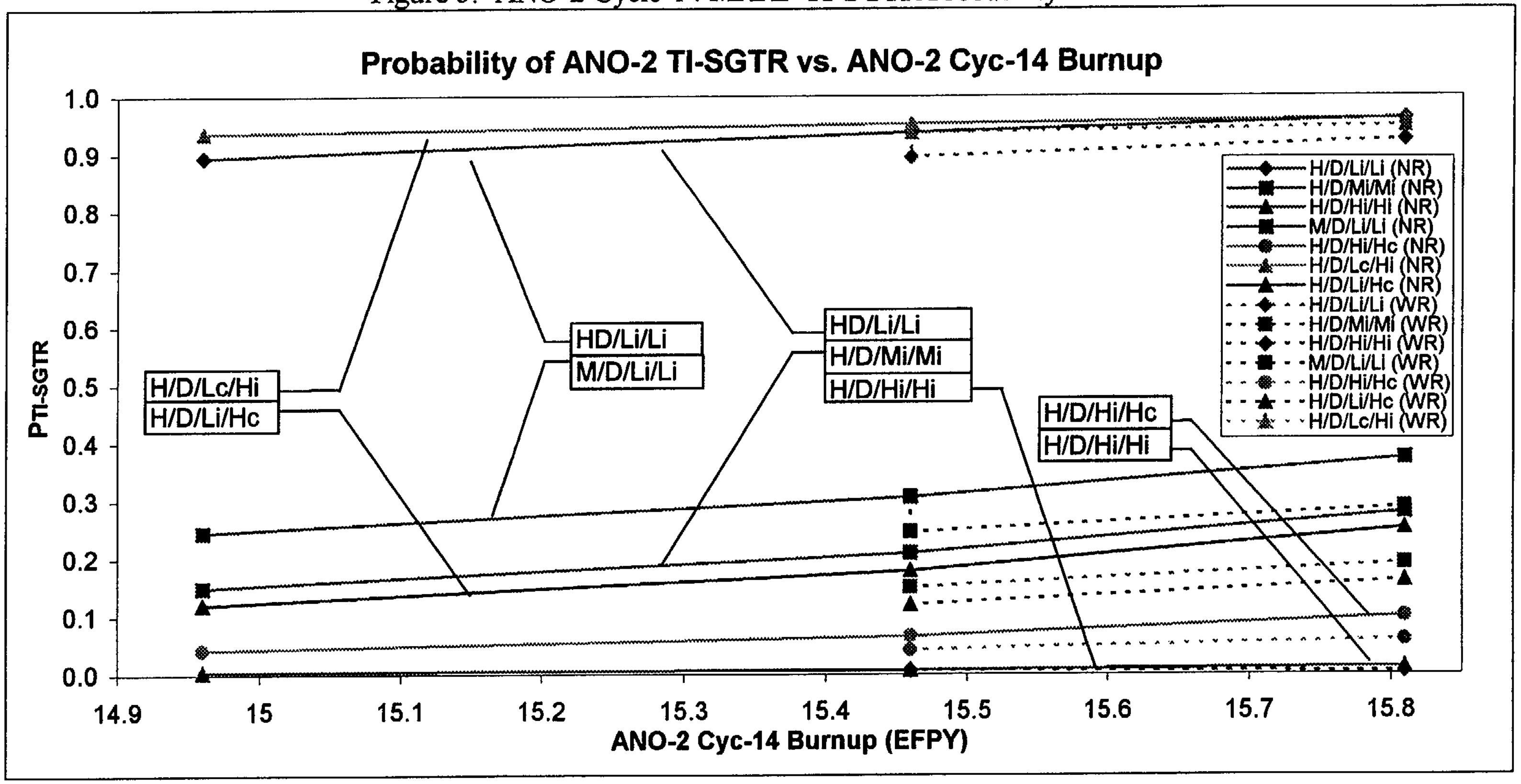
[2] Beginning, Middle, and End of Period (BOP, MOP, EOP) are defined as follows:
BOP: 14.96 EFPY (2P99)
MOP: 15.46 EFPY (proposed 2P00)
EOP: 15.81 EFPY (scheduled 2R14)

[3] Case 1 and 4 results (2/2 SGs) are estimated from Case 7 and 8 (1/2 SGs) results via Boolean doubling of the 1/2SG results, i.e. $(1-(1-(1/2\ SG))^2)$

[4] MAAP and PROBFAIL calculations were not performed for Cases 10, 11, 13, 14, 15, 16, and 17; these results are estimated using Cases 1 through 5 and Case 12 results.

[5] Case 12 results are conservatively assumed to be the maximum of Cases 5 and 6.

Figure 5. ANO-2 Cycle-14 MAAP TI-SGTR Probability Results



4.4.2 ANO-2 Core Damage Accidents Vulnerable to TI-SGTR

A review of the ANO-2 Rev-1 core damage results documented in Ref. 7 was performed in order to identify which ANO-2 severe accidents are expected to lead to a TI-SGTR event. Section 4.4.2.1 documents this process and its results, i.e., the ANO-2 severe accident sequences, their frequencies, and the overall frequency of this condition at ANO-2. Since the ANO-2 Rev-1 risk assessment did not account for the SG pressure condition, nor did it assess the potential for RCS depressurization or for the potential for losing an RCP loop seal during a severe accident, an assessment of these issues was performed. This effort and its results are documented in Section 4.4.2.2. The results of Sections 4.4.2.1 and 4.4.2.2 are combined in Section 4.4.2.3 to produce an estimate of the frequency of ANO-2 severe accident sequences involving any combination of High, Medium, or Low RCS pressure, with Dry or Wet SG conditions, with High, Medium, or Low SG pressure, and with either intact or cleared RCP loop seals.

4.4.2.1 Assessment of ANO-2 Core Damage Accidents Involving of Probability of High RCS Pressure and Dry SG Condition

Table 8, below, provides a summary of the ANO-2 PSA Rev-1 core damage results from Table 5 of Ref. 7. Attachment C provides a description of the ANO-2 accident sequence nomenclature excerpted from Ref. 7. It should be noted that the ATWS and Internal Flooding CDF results are based on scoping analyses and, as such, are conservative upper bound CDF estimates.

Table 8. ANO-2 Rev-1 Core Damage Frequency Results

Accident Sequence	Core Damage Frequency (/rx-yr)
AU	4.061E-07
AX	4.363E-08
MU	2.600E-07
MX	4.898E-07
SBF	6.127E-08
SBU	1.144E-09
SBX	6.456E-10
SU	1.199E-06
SX	4.313E-07
RBF	1.150E-07
RBU	5.275E-09
RBX	1.937E-09
RU	5.885E-12
RX	1.768E-08
TBF	1.511E-05
TBX	2.667E-06
TQBF	7.211E-09
TQBU	1.215E-10
TQBX	4.321E-10
TQU	3.538E-10
TQX	2.173E-08
Internal Events	**2.084E-05**

Accident Sequence	Core Damage Frequency (/rx-yr)
ISLOCA	3.27E-07
ATWS	1.59E-06
Internal Flooding	<1.0E-06
Total Frequency	**2.28E-05**

The above results are based on the application of one or, in some cases, two operator recoveries per cutset. Although this approach simplified the assessment of operator action dependencies, this approach produces conservatively high CDF results. One significant result of this approach was that the Loss of DC power-initiated cutsets dominated the ANO-2 Rev-1 CDF estimate. A review of the cutsets involving this particular initiator revealed that additional operator recoveries are available. This review involved a walkdown of the plant, discussions with operators, discussions with System Engineers, and the execution of the loss of DC event on the ANO-2 plant simulator with operators responding to the event. Attachment D describes this review and provides recommended operator recovery actions and failure probabilities for these combined operator recovery actions accounting for their interdependencies. Note that other discussions in Attachment D, namely those related to SG pressure control and RCP seal loop have been superceded by analyses presented in Attachment G. A detailed description of the process used to review and apply these additional operator recoveries to the Loss of DC initiated ANO-2 Rev-1 cutsets is provided in Attachment E. As noted in this attachment, the recoveries were applied via the use of Human Recovery Rule (HRA) post-processing rule files. The rule files are implemented via the QRECOVER code, which is part of the EPRI Risk and Reliability Workstation, in a manner consistent with that done as part of the development of the ANO-2 Rev-1 PSA results.

The Attachment D investigation also provided recommendations on the use of a generic operator recovery, OA2. This recovery action was applied to the ANO-2 Rev-1 cutset results using a modified version of the ANO-2 Rev-1 HRA rules. The development of these rule files are documented in Attachment D of Ref. 10. The ANO-2 Rev-1 HRA rule file identifies where additional operator actions could be applied to the ANO-2 Rev-1 cutset results. The modified version of this file applies OA2 where additional operators are expected. The OA2 rule file, i.e., HRArules.txt, is provided in Attachment A.

Application of the additional Loss of DC Power and generic second operator recovery action revised the ANO-2 Core Damage Frequency results, which are shown in Table 9, below. These results are based on cutset file A2R1COM9.CUT. This file, along with all others used or generated in the assessment and application of the Loss of DC and generic second recovery events are noted in Attachment D, are listed in Attachment A, and are provided as part of this calculation.

Table 9. ANO-2 Rev-1 Core Damage Frequency Results*

Accident Sequence	Core Damage Frequency (/rx-yr)
AU	4.061E-07
AX	4.361E-08

Accident Sequence	Core Damage Frequency (/rx-yr)
MU	7.487E-08
MX	4.413E-07
SBF	5.548E-08
SBU	2.041E-10
SBX	9.737E-11
SU	1.199E-06
SX	4.252E-07
RBF	1.011E-07
RBU	9.111E-10
RBX	6.430E-10
RU	5.878E-12
RX	1.749E-08
TBF	6.167E-06
TBX	2.228E-06
TQBF	2.146E-10
TQBU	8.384E-12
TQBX	5.050E-13
TQU	2.563E-10
TQX	2.130E-08
Internal Events	**1.118E-05**
ISLOCA	3.27E-07
ATWS	1.59E-06
Internal Flooding	<1.0E-06
Total Frequency	**1.410E-05**

* based on cutset a2r1com9.cut.

The revised ANO-2 Rev-1 accident sequences were reviewed and categorized according to vulnerability to TI-SGTR. Of the parameters relevant to TI-SGTR noted in Section 4.4.1, only two are indigenous to the ANO-2 cutset results. These are :

(1) RCS pressure condition (High, Medium, or Low) and
(2) SG inventory condition (Wet or Dry).

Other relevant parameters were either addressed in the ANO-2-specific MAAP analyses or will require additional assessment. Classification of the ANO-2 internal events accident sequences according to RCS pressure and SG inventory was done using the ANO-2 Rev-1 event tree logic in Sections 3.1.2.3 through 3.1.2.6 of Ref. 11 and using the ANO-2-specific MAAP accident sequence results documented in Ref. 12. The ANO-2 internal events sequence classifications (excluding the "special events" ISLOCA, ATWS, and Internal Flooding) are as follows:

Large Break LOCA Sequences (AU, AX):

RCS Pressure: The RCS pressure at the time of core damage for accidents initiated by the Large Break LOCA (LBLOCA), i.e., initiator A, is assumed to be "**low**", i.e., less than 100 psia per Ref. 12. The pipe break size assumed in the Ref. 12 MAAP LBLOCA analysis was 12" in diameter. Although this is greater than the 4.3" minimum LBLOCA break size defined by Ref. 11, this fact is not expected to change the RCS pressure category for LBLOCAs.

SG Inventory: The SG is expected to be **wet** for this accident.

Medium Break LOCA Sequences (MU, MX):
RCS Pressure: The RCS pressure at the time of core damage for accidents initiated by the Medium Break LOCA (MBLOCA), i.e., initiator M, is assumed to be "**low**", about 150 psia per Ref. 12. The pipe break size assumed in the Ref. 12 MAAP MBLOCA analysis was 1.9" in diameter. Although this is less than the minimum MBLOCA break size defined by Ref. 11 (i.e., 1.9" to 4.3"), this fact is judged not to change the RCS pressure category for MBLOCAs. The basis for this judgment is that Ref. 11 defined the minimum MBLOCA break size as large enough to assure that the RCS will depressurize to the HPSI shutoff head due to the break relief. Per the Ref. 12 MAAP results indicate that significant core damage is long delayed (about 3 hours after the initiator); thus, RCS depressurization to just above the LPSI shutoff head is expected and is consistent with the ANO-2 Rev-1 analysis assumptions.
SG Inventory: The SGs are expected to be **wet** at the time of core damage.

Small Break LOCA Sequences without Secondary Heat Removal (SBF, SBU, SBX):
RCS Pressure: The RCS pressure at the time of core damage for accidents initiated by the Small Break LOCA (SBLOCA), i.e., initiator S, without secondary heat removal is assumed to be "**high**", i.e., conservatively assumed to be at the RCS safety relief valve setpoint of 2500 psia. A MAAP analysis for sequences of this type was not performed in Ref. 12.
SG Inventory: The SGs are expected to be **dry** at the time of core damage for this accident.

Small Break LOCA Sequences with Secondary Heat Removal (SU, SX):
RCS Pressure: The RCS pressure at the time of core damage for accidents initiated by the Small Break LOCA, i.e., initiator S, with secondary heat removal is assumed to be "**medium**", i.e., about 1100 psia, since secondary heat removal is available via the SGs, which will be pressurized no higher than the lowest MSSV setpoint pressure of ~1100 psia. The Ref. 12 MAAP results confirm this assumption.
SG Inventory: The SGs are expected to be **wet** at the time of core damage for this accident.

Spontaneous SGTR Sequences (RU, RX, RBF, RBU, RBX):
These sequences involve SGTR as an accident initiator, R; thus, these accident sequences are classified as Large Early Release events independent of RCS pressure or SG inventory at the time of core damage.

Transient-Initiated Sequences without Secondary Heat Removal (TBF, TBX):
RCS Pressure: The RCS pressure at the time of core damage for transient accidents without secondary heat removal, i.e., TBF and TBX, is assumed to be "**high**", i.e., at the RCS relief valve setpoint of 2500 psia, since secondary heat removal is not available and since no HPSI makeup occurs in these sequences. Core damage is assumed to occur either because RCS depressurization capability is not available or because HPSI injection capability is not available.
SG Inventory: The SG is assumed to be **dry** at the time of core damage for this accidents.

Transient-Initiated LOCA Sequences with Secondary Heat Removal (TQU, TQX):
RCS Pressure: The RCS pressure at the time of core damage for accidents involving a transient-induced LOCA, i.e., events TQ, with secondary heat removal is assumed to be "**medium**", i.e.,

about 1100 psia, since secondary heat removal is available via the SGs, which will be pressurized no higher than the lowest MSSV setpoint pressure of ~1100 psia.
SG Inventory: The SGs are expected to be **wet** at the time of core damage for this accident.

Transient-Initiated LOCA Sequences without Secondary Heat Removal (TQBF, TQBU, TQBX):
RCS Pressure: The RCS pressure at the time of core damage for accidents involving a transient-induced LOCA, i.e., events TQ, without secondary heat removal is assumed to be "**high**", i.e., conservatively assumed to be at the RCS safety relief valve setpoint of 2500 psia. A MAAP analysis for sequences of this type was not performed in Ref. 12.
SG Inventory: The SGs are expected to be **dry** at the time of core damage for this accident.

Table 10 summarizes the above discussions on the RCS pressure and SG inventory conditions associated with the ANO-2 core damage sequences.

Table 10. RCS Pressure/SG Inventory Classification for ANO-2 Internal Events Accident Sequences (excluding special events)

Accident Sequence	Core Damage Frequency (/rx-yr)	RCS Pressure	SG Inventory
AU	4.061E-07	Low	Wet
AX	4.361E-08	Low	Wet
MU	7.487E-08	Low	Wet
MX	4.413E-07	Low	Wet
SBF	5.548E-08	High	Dry
SBU	2.041E-10	High	Dry
SBX	9.737E-11	High	Dry
SU	1.199E-06	Medium	Wet
SX	4.252E-07	Medium	Wet
RBF	1.011E-07	N/A	N/A
RBU	9.111E-10	N/A	N/A
RBX	6.430E-10	N/A	N/A
RU	5.878E-12	N/A	N/A
RX	1.749E-08	N/A	N/A
TBF	6.167E-06	High	Dry
TBX	2.228E-06	High	Dry
TQBF	2.146E-10	High	Dry
TQBU	8.384E-12	High	Dry
TQBX	5.050E-13	High	Dry
TQU	2.563E-10	Medium	Wet
TQX	2.130E-08	Medium	Wet
Internal Events	**1.118E-05**		

Table 11 condenses the results presented in Table 10 by summarizing the internal events CDF associated with each RCS pressure and SG inventory classifications. This process was performed in the EXCEL spreadsheet HML-WD.xls, which is provided in Attachment A.

Table 11. RCS Pressure/SG Inventory Internal Events Classification Frequencies

RCS Pressure/SG Inventory	CDF (/rx-yr)	Fraction of Total Internal Events CDF (excluding SGTR)
High/Wet	0.000E+00	0.000
Medium/Wet	1.646E-06	0.149
Low/Wet	9.659E-07	0.087
High/Dry	8.452E-06	0.764
Medium/Dry	0.000E+00	0.000
Low/Dry	0.000E+00	0.000
SGTR	1.201E-07	N/A
Internal Events Total	**1.118E-05**	**1.000**

The CDFs associated with several "Special Events" were assessed as part of the ANO-2 PSA risk assessment. These special events were the Interfacing Systems LOCA (ISLOCA), Anticipated Transient Without Scram (ATWS), and Internal Flooding accidents. The ISLOCA and ATWS risk analyses were performed as "screening" analyses not part of the internal events PSA risk assessment. Although these two risk assessments produced quantitative CDF estimates, these estimates are conservatively high. The Internal Flooding analysis is a "vulnerability" analysis; no cutsets were generated in this assessment. The CDF estimate generated in this analysis is considered an upper bound CDF estimate for the Internal Flooding.

The ISLOCA accident is a containment bypass event and, as such, is defined as a LERF event. Therefore, since it is a LERF event and the occurrence of a TI-SGTR cannot worsen its LERF contribution, further assessment of the relative frequencies of its RCS pressure and SG inventory condition "split fractions" is not necessary.

The ATWS event represents a significant pressure challenge to the RCS and SGs. In addition, the documentation of the ATWS analysis [Ref. 13] provides information regarding the condition of SG inventory at the time of core damage. Using event tree logic documented in Figures 1, 2, and 3 and Table 3 of Ref. 13, the ATWS core damage sequences were classified according to RCS pressure and SG inventory conditions was performed. Based on the ATWS event tree logic, the Turbine Trip (T1) core damage sequences 27, 28, 54, and 55, the Loss of PCS (T2) core damage sequences 15, 16, 18, 19, 33 ,34, 36, and 37, and the Loss of Offsite Power core damage sequences 15, 16, 18, 19, 20, 34, 35, 37, 38, and 39 were assumed to involve High RCS pressure with dry SG conditions and all other ATWS core damage sequences were assumed to involve High RCS pressure with wet SG conditions. For the ATWS sequences, since the RCS pressure is likely to be very high, on the order of 3700 psia, regardless of SG materiel condition, all High/Dry sequences were assumed to lead to TI-SGTR i.e., TI-SGTR probability is 1.0); and, the High/Wet sequences were assumed not to lead to TI-SGTR (i.e., TI-SGTR probability is 0). This assumption results in an increase in the base LERF at all burnup points, but no change in LERF as a function of burnup.

Table 12 summarizes the ATWS CDFs associated with each RCS pressure and SG inventory classification.

Table 12. RCS Pressure/SG Inventory ATWS Classification Frequency

RCS Pressure/ SG Inventory	CDF (/rx-yr)	Fraction of ATWS CDF
High/Wet	1.405E-06	0.884
Medium/Wet	0	0
Low/Wet	0	0
High/Dry	1.847E-07	0.116
Medium/Dry	0	0
Low/Dry	0	0
ATWS Total	**1.590E-06**	**1.000**

Although an ANO-2 external events risk analysis is not currently available, in an effort to assure completeness of the TI-SGTR risk issue, a review of the risks from external events at ANO-2 were investigated. A description of this investigation is provided in Attachment F. The overall results of this investigation were the following:

- The qualitative IPEEE results for external events can be accounted for as an add-on 50% contribution to the estimate of internal events.
- The feedwater supply availability is not expected to be more significantly affected by external events than internal events. Thus the fraction of core damage events that lead to SGTR risk should not be much higher for external events than for internal events.

Both the Internal Flooding and the External Events accidents are considered external events. In fact, as a result of the simplistic treatment of external events, there is likely some double-counting of these Internal Flooding risk contributor. Based on the above results, it was assumed that the Internal Events CDF "split fractions" on RCS pressure and SG inventory condition apply to both the Internal Flooding and External Events accidents. These "split fractions" are the "Fraction of Total Internal Events CDF (excluding SGTR)" listed in Table 11, above. Applying this assumption to the overall Internal Flooding and External Events CDF results, i.e., yields the Table 13 and 14 results, below.

Table 13. RCS Pressure/SG Inventory Internal Flooding Classification Frequencies

RCS Pressure/ SG Inventory	CDF (/rx-yr)	Fraction of Internal Flooding CDF
High/Wet	0.000E+00	0.000
Medium/Wet	1.488E-07	0.149
Low/Wet	8.731E-08	0.087
High/Dry	7.639E-07	0.764
Medium/Dry	0.000E+00	0.000
Low/Dry	0.000E+00	0.000
Internal Flooding Total	**1.00E-06**	**1.000**

Table 14. RCS Pressure/SG Inventory External Events Classification Frequency

RCS Pressure/ SG Inventory	CDF (/rx-yr)	Fraction of External Events CDF
High/Wet	0.000E+00	0.000
Medium/Wet	8.318E-07	0.149
Low/Wet	4.882E-07	0.087
High/Dry	4.272E-06	0.764
Medium/Dry	0.000E+00	0.000
Low/Dry	0.000E+00	0.000
External Events Total	**5.592E-06**	**1.000**

Table 15, below, summarizes the RCS pressure and SG inventory classification results from Tables 11, 12, 13, and 14 for Internal Events, ATWS, Internal Flooding, and External Events, respectively.

Table 15. RCS Pressure/SG Inventory ANO-2 Total Core Damage Classification Frequency

RCS Pressure/ SG Inventory	CDF (/rx-yr)	Fraction of Total CDF
High/Wet	1.405E-06	0.071
Medium/Wet	2.626E-06	0.133
Low/Wet	1.541E-06	0.078
High/Dry	1.367E-05	0.694
Medium/Dry	0.000E+00	0.000
Low/Dry	0.000E+00	0.000
SGTR	1.201E-07	0.006
ISLOCA	3.270E-07	0.017
Total Core Damage	**1.969E-05**	**1.000**

An additional sorting of the ANO-2 CDF results was done to separate accidents which lead to a Large Early Release for reasons other than a TI-SGTR. Since these accidents are already LERF events, it is inappropriate to double count them as TI-SGTR LERF contributors. The ANO-2 IPE Level-2 results documented in Ref. 3 can be used to estimate the fraction of the ANO-2 Rev-1 CDF that contributes the "non-TI-SGTR" LERF. Conservatively assuming that the spontaneous SGTR and the ISLOCA accidents are LERF events, the remaining "non-TI-SGTR" LERF contributors are due to containment failure. Per Table 4.7-4 of the ANO-2 IPE provided in Ref. 3, excluding the spontaneous SGTR and the ISLOCA accidents, Plant Damage States (PDSs) D2-R, D4-R, E2-R, E4-R, and E6-R are Large Releases. These PDSs are considered LERF events. Table 16, below, lists the frequency of these PDSs and the total PDS-based CDF taken directly from Table 4.6-2 provided in Ref. 3. The fraction of PDSs leading to a LERF independent of a TI-SGTR event is the sum of these PDS frequencies divided by the PDS CDF. This fraction, called the Baseline LERF Fraction, is provided in Table 16, below.

Table 16. Calculation of "Non-TI-SGTR" LERF

LERF PDS	LERF (/rx-yr)
D2-R	3.20E-08
D4-R	1.36E-06
E2-R	1.12E-09
E4-R	2.56E-07
E6-R	5.19E-07
Sum of PDS CDF Leading to a LERF (excluding Spontaneous SGTRs and ISLOCAs)	2.17E-06
Sum of PDS CDF	3.71E-05
Fraction of PDSs Leading to a LERF (excluding Spontaneous SGTRs and ISLOCAs)	**5.84E-02**

The Baseline LERF Fraction can be used to calculate the Baseline LERF value, a value which will used later in the analysis:

ANO-2 Baseline LERF = (Baseline LERF Fraction) * (Total ANO-2 CDF – SpSGTR CDF – ISLOCA CDF) + (SpSGTR CDF) + (ISLOCA CDF)
= (0.0584) * (1.969E-05 - 1.201E-07 - 3.270E-07) + (1.201E-07)n + (3.270E-07)
= 1.572E-06/rx-yr

Since Table 15 indicates that the Medium/Dry and the Low/Dry conditions are null contributors to TI-SGTRs, only High/Dry condition need be considered further as a contributor to TI-SGTR. Thus, the frequency of ANO-2 core damage events which are vulnerable to TI-SGTR are the product of the ANO-2 core damage frequency associated with a High/Dry condition and the compliment of the Baseline LERF Fraction, i.e.,

TI-SGTR CDF Vulnerable to TI-SGTR = (High/Dry CDF) * (1 – Baseline LERF Fraction)
= (1.367E-05/rx-yr) * (1 – 0.0584)
= 1.287E-05/rx-yr

In summary, for ANO-2, the events vulnerable to a TI-SGTR are core damage events involving a High RCS/Dry SG condition and these core damage events exclude core damage events which lead to a Large Early Release for reasons other than a TI-SGTR. The frequency of such CDFs is 1.287E-06/rx-yr. It represents approximately 65% of the total ANO-2 CDF of 1.969E-05/rx-yr.

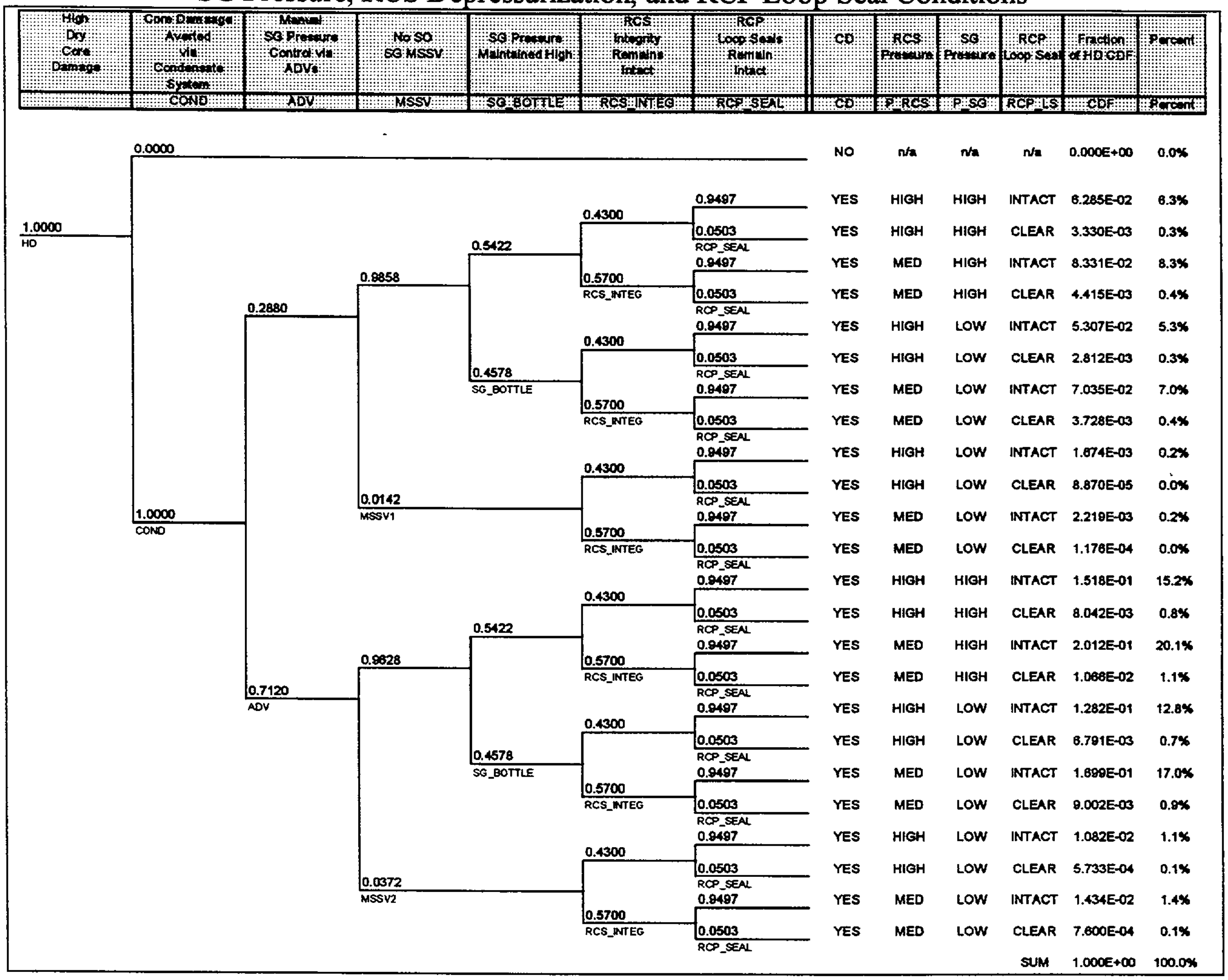
4.4.2.2 Assessment of SG Pressure, RCS Depressurization, and Loss of RCP Loop Seal

The frequency of ANO-2 core damage events vulnerable to a TI-SGTR calculated in Section 4.4.2.1, above, does not account for several additional factors that affect the vulnerability of the SG tubes to TI-SGTR. These are

(1) the SG pressure condition during the core damage progression,
(2) the potential for an RCS depressurization during the accident progression, and
(3) the possibility of losing a RCP loop seal.

An event tree, shown in Figure 6, was developed to account for these issues. The basis and values of the branching probabilities in the event tree are presented below.

Figure 6. Event Tree for Assessing Additional TI-SGTR Issues:
SG Pressure, RCS Depressurization, and RCP Loop Seal Conditions



As noted above, since "wet" SG conditions are not vulnerable to TI-SGTR, since the only non-zero "dry" SG condition involves a high RCS pressure, and since these events must not already be LERF events, only the non-LERF high RCS pressure/dry SG (H/D) core damage category requires additional analysis using the subject event tree. Per Section 4.4.2.1, the frequency of the entry condition is 1.287E-05/rx-yr. A description of the event tree top events, the probability of each of these events, and the basis for these values are presented in Table 17, below.

The first event, *COND*, accounts for the potential for averting core damage via the use of the Condensate system. This event was included because the ANO-2 Rev-1 core damage assessment did not credit the use of the Condensate system as a means of SG makeup. The system requires the SGs to be depressurized to approximately 550 psi, before it can be used. Conservatively, no credit was taken for this action in this analysis.

The next three top events, *ADV, MSSV*, and *SG_BOTTLE*, account for the SG pressure condition during the accident progression. As noted above, the SG pressure condition was not accounted for in the ANO-2 Rev-1 core damage risk analysis. Event ADV accounts for the operator use of the ADVs to control SG pressure before it dries out. The failure probability for event ADV was assessed by Boolean summing ADV equipment failure and operator failure to use the system. A review of the ANO-2 accident cutset results was performed in order to assess the fraction of HD CDF that involved the failure of control or motive power to any ADV valve. This fraction was taken to be the equipment failure probability. Per Attachment G, operator failure was assessed to be 0.0466. This action is directed by the ANO-2 EOPs and there is ample time for the action. Use of the ADVs minimizes the cycling of the MSSVs and therefore minimizes the likelihood that an MSSV will fail to reclose (i.e., stick open), Event MSSV. The probability of event MSSV depends on the success/failure of event ADV. If ADV fails, it was assumed that there are 85 MSSV open demands in the HD accident. This number of demands was generated in Attachment H for a Station Blackout (SBO) event at ANO-2. As such, this count is expected to bound all other accidents. If event ADV is successful, the number of MSSV open demands is reduced. It was arbitrarily assumed that only 25 MSSV demands would occur prior to the use of the ADVs. This count is probably quite conservative. Event SG_BOTTLE represents the failure to completely "bottle" both SGs, i.e., isolate all but the ADV path, prior to the SGs dry. It assumes that the probability that the support systems required to perform the valve isolations are accounted for in the ADV support system assessment. This assumption takes no credit for manual actions to close the subject isolation valves. The event also accounts for the operator action to perform the action. And, it accounts for the potential for a SG leak large enough to fully depressurize either SG. Per Attachment G, operator failure associated with this event was assessed to be 0.1393.

Event *RCS_INTEG* accounts for the loss of RCS integrity due to stuck open primary Safety Relief Valve (SRV) or an intentional RCS depressurization by an operator. The success of this event reduces the RCS pressure, reduces the transport of heat to the SGs, and reduces the pressure difference across the SG tubes. Although the event is likely to fully depressurize the RCS, it was conservatively assumed that the event depressurizes the RCS to a "Medium" pressure, i.e. to about 1400 psia. At "Medium" RCS pressures, although the potential for a TI-SGTR is greatly reduced, this potential is significantly higher than that at the "Low" RCS pressure, i.e., the RCS was fully depressurized. The probability of a SRV sticking open was taken from NUREG-1570, Ref. 1. It is noted that no credit for operator intentional RCS depressurization was taken for this event.

Event *RCP_SEAL* accounts for the potential that a RCP loop seal is lost. The loss of a loop seal increases the potential for a TI-SGTR by increasing the potential for a unidirectional natural convention flow between the degrading core and the relatively cool SG in the affected RCP loop. This event accounts for both mechanical failure of a RCP seal and operator action to "bump" a RCP pump in order to intentionally clear the loop seal. The ANO-2 RCPs use the multistage N-9000 seal, which has been designed to accommodate station blackout conditions and has been demonstrated to be robust to a wide range of upsets including operation without seal cooling. Thus, a low failure probability (0.01) was assigned to the mechanical failure part of this event.

Although this value is arbitrary, it is believed to be conservatively high. The probability of an intentional operator action to clear a RCP loop is governed by the ANO-2 Severe Accident Management Guidelines (SAMGs). Per these guidelines, it is appropriate to clear the seals only if the associated SG is being fed. Thus, the probability of an operator inappropriately clearing a loop seal when feedwater is not available is low. Per Attachment G, operator failure was assessed to be 0.0407.

Table 17. Branching Logic for Event Tree for Assessing Additional TI-SGTR Issues

Top Event	**Description**	**Failure Probability**	**Basis**
COND	Condensate Not Available	1.0	No credit taken for use of the Condensate pumps as a means of maintaining SG inventory
ADV	ADVs not used to control SG pressure	0.7120	Boolean sum of ADV-EQPT and ADV-OA
ADV-EQPT Equipment Unavailability	ADVs Unavailable due to either support system failure or ADV valve failure	0.6980	Based on Cutset review. Supporting calculations associated with this value are discussed in spreadsheet TISGTR-2P99.xls and provided in files contained in ADV.zip in Attachment A.
ADV-OA Operator Action Failure	Operator fails to use ADVs to control SG pressures to minimize MSSV open demands	0.0466	Estimated mean value. Assumes operator fails to follow EOPs. See Attachment G
MSSV	MSSV on either SG sticks open depressurizing the SG		
MSSV1	MSSV on either SG sticks open after a demand, given use of ADVs to minimize number of MSSV demands	0.0142	Assumes 25 MSSV cycles, given ADVs used to control SG pressure. Number of cycles was an arbitrary fraction of the number of cycles for the case in which ADVs are not used. The MSSV valve failure rates were based on Att. H): λ = 4.50E-03/demand (1st demand) λ = 3.93E-04/demand (subsequent demands)
MSSV2	MSSV on either SG sticks open after a demand, given <u>no</u> use of ADVs to minimize number of MSSV demands	0.0372	Based on an estimated 85 MSSV cycles, given ADVs not used to control SG pressure. Number of cycles was estimated in Att. H. The MSSV valve failure rates were based on Att. H): λ = 4.50E-03/demand (1^{st} demand) λ = 3.93E-04/demand (subsequent demands)
SG_BOTTLE	Isolation of the SGs to assure that they remain pressurized during the core damage progression	0.4578	Boolean sum of SGBOT-LEAK and SGBOT-OA
SGBOT-LEAK	SG Isolation Valves Fail to Operate or Leak	0.37	In NUREG-1570, for the Surry evaluation, it was assumed that once isolated that the probability of one of its three SG to

Top Event	Description	Failure Probability	Basis
			experience a leak was 0.5. For a single SG, this translates to a leakage probability of about 0.21. Thus, for ANO-2 with two SGs, the probability of a leakage is the Boolean sum of this probability, i.e., 0.37. Note that the Staff defined separate release categories for MSSV stuck open releases and those lower releases were associated with MSSV leakage. No credit for these lower releases was taken in this analysis. Valve failure to close was neglected, since it is much smaller than the assessed leakage probability.
SGBOT-OA	Oper Fails to Attempt to Bottle SGs	0.1393	Estimated value. Assumes operator fails to follow EOPs. See Attachment G
RCS_INTEG	Loss of RCS integrity due to stuck open primary Safety Relief Valve (SRV) or operator intentional depressurization	0.5700	Boolean sum of PSRVSO and RCS-OA
PSRVSO (Equipment Failure)	Loss of RCS integrity due to stuck open (SO) primary Safety Relief Valve (PSRV)	0.5700	Assumes 0.14 probability of a PSRV SO prior to core uncovery (per NUREG-1570) and Assumes 0.5 probability of PSRV SO after core uncovery (per NUREG-1570). These probabilities are summed in a Boolean manner.
RCS-OA (Operator Action)	Operator intentionally depressurizes RCS w/o RCS makeup available	0	Estimated value. Conservatively assumed to not occur.
RCP_SEAL	RCP loop seal remains intact during core damage progression	0.0503	Boolean sum of RCP-SL and RCP-OA
RCP-SL Equipment Failure	RCP Seal LOCA occurs	0.0100	Estimated value. ANO-2 N-9000 RCP seal design is expected to remain intact.
RCP-OA (Operator Action)	Operator Intentionally clears a RCP loop seal via a RCP pump "bump"	0.0407	Estimated mean value. SAMGs recommend against this action, unless feed is provided to associated SG. See Attachment G

Application of the Figure 3, TI-SGTR event tree, using the Table 17 branching probabilities yields the results shown in Table 18, below. These results provide the fraction of HD core damage frequency associated with each of eight TI-SGTR vulnerability states. Each of these states are a combination of RCS pressure/SG inventory/SG pressure/RCP Loop seal conditions. In the next section, these fractions are combined with the HD CDF value provided in Table 15 and each endstate will be combined with the probability of a TI-SGTR provided in Table 18, below.

Table 18. Summary of TI-SGTR Vulnerabilities

CD	RCS Pressure Condition	SG Inventory Condition	SG Pressure Condition	RCP Loop Seal Condition	Fraction of HD CD
No CD	n/a	n/a	n/a	n/a	0.000E+00
CD	HIGH	DRY	HIGH	INTACT	2.146E-01
CD	HIGH	DRY	HIGH	CLEAR	1.137E-02
CD	HIGH	DRY	LOW	INTACT	1.937E-01
CD	HIGH	DRY	LOW	CLEAR	1.027E-02
CD	MED	DRY	HIGH	INTACT	2.845E-01
CD	MED	DRY	HIGH	CLEAR	1.508E-02
CD	MED	DRY	LOW	INTACT	2.568E-01
CD	MED	DRY	LOW	CLEAR	1.361E-02
				SUM	1.000E+00

4.4.2.3 Frequency of ANO-2 Core Damage Accidents Vulnerable to TI-SGTR

Applying these split fractions to the HD CDFs which are not LERF due to reasons other than TI-SGTR produces the HD CDFs vulnerable to TI-SGTR. The result of this combination is provided in Table 19, below.

Table 19. Summary of TI-SGTR Vulnerability CDFs

CD	RCS Pressure Condition	SG Inventory Condition	SG Pressure Condition	RCP Loop Seal Condition	Non-LERF HD CDF (/rx-yr)
No CD	n/a	n/a	n/a	n/a	0.000E+00
CD	HIGH	DRY	HIGH	INTACT	2.763E-06
CD	HIGH	DRY	HIGH	CLEAR	1.464E-07
CD	HIGH	DRY	LOW	INTACT	2.494E-06
CD	HIGH	DRY	LOW	CLEAR	1.321E-07
CD	MED	DRY	HIGH	INTACT	3.662E-06
CD	MED	DRY	HIGH	CLEAR	1.941E-07
CD	MED	DRY	LOW	INTACT	3.306E-06
CD	MED	DRY	LOW	CLEAR	1.752E-07
				SUM	1.287E-05

4.4.3 Assessment of ANO-2 Risk Due to TI-SGTR

At a given point in time (or burnup), the LERF associated with TI-SGTRs ($LERF_{TI\text{-}SGTR}$) is the sum of the products of

(1) the frequency (F_i) of each category of core damage events vulnerable to TI-SGTR and

(2) its associated TI-SGTR probability (P_i).

That is,

$$LERF_{TI-SGTIF} = \sum_{i}^{n} F_i P_i,$$

where, n is the number of TI-SGTR vulnerable core damage categories.

For ANO-2, as noted above, the only core damage accidents vulnerable to TI-SGTR is the High RCS Pressure/Dry SG Pressure (i.e., H/D) core damage event. And, per Table 19, after considering the potential for depressurization of the RCS during the severe accident progression, and the possible SG pressure and RCP loop seal conditions, conservative analysis resulted in eight core damage categories vulnerable to TI-SGTR. These TI-SGTR categories and the frequency associated with each, are shown in Table 19. The probability of TI-SGTR associated with each of these categories is provided in Table 7 at several burnup points during the latter half of ANO-2 Cycle-14. These burnup points are:

(1) Immediately after the 2P99 SG inspection/repair campaign at 14.96 EFPY (called the "Beginning of Period (BOP)"),
(2) Immediately before a proposed planned SG inspection/repair campaign 2P00 at 15.46 EFPY (called "Middle of Period, with No 2P00 SG Repair (MOP-NR)"),
(3) Immediately before the scheduled Refueling outage at the End of Cycle 14 at 15.81 EFPY (called the "End of Period, with No 2P00 Repair (EOP-NR)"),
(4) Immediately after the proposed 2P00 SG inspection/repair campaign at 15.46 EFPY (called "Middle of Period With 2P00 SG Repair (MOP-WR)"), and
(5) Immediately before the scheduled 2R14 Refueling outage at the End of Cycle 14 at 15.81 EFPY (called the "End of Period, With 2P00 SG Repair (EOP-WR)").

Note that the TI-SGTR probability just after 2P00 with repairs was assumed to be the same as that just after 2P99. The TI-SGTR probability just before 2R14 w/ a 2P00 repair outage is estimated by assuming that the TI-SGTR probability increases at the same rate per EFPY as it does between 2P99 to 2P00. The results of Tables 7 and 19 are compiled into Table 20, below.

Table 20.

TI-SGTR Vulnerability Category	**Frequency (/rx-yr)**	**TI-SGTR Frequency 2P99 After Repairs (14.96 EFPD)**	**TI-SGTR Frequency 2P00 No Repairs (15.46 EFPD)**	**TI-SGTR Frequency 2R14 No Repairs (15.81 EFPD)**	**TI-SGTR Frequency 2P00 With Repairs (15.46 EFPD)**	**TI-SGTR Frequency 2R14 With Repairs (15.81 EFPD)**
H/D/Hi	2.763E-06	3.964E-03	7.088E-03	1.229E-02	3.964E-03	6.151E-03
H/D/Hc	1.464E-07	4.144E-03	7.204E-03	1.227E-02	4.144E-03	6.304E-03
H/D/Li	2.494E-06	8.951E-01	9.373E-01	9.616E-01	8.951E-01	9.246E-01
H/D/Lc	1.321E-07	9.358E-01	9.526E-01	9.602E-01	9.358E-01	9.476E-01
M/D/Hi	3.662E-06	1.086E-03	2.321E-03	4.772E-03	1.086E-03	1.919E-03
M/D/Hc	1.941E-07	1.135E-03	2.359E-03	4.765E-03	1.135E-03	1.966E-03
M/D/Li	3.306E-06	2.453E-01	3.069E-01	3.733E-01	2.453E-01	2.884E-01
M/D/Lc	1.752E-07	2.564E-01	3.119E-01	3.728E-01	2.564E-01	2.956E-01
SUM	**1.287E-05**					

Using the information in this table, the intermediate results of applying the TI-SGTR LERF equation is provided in Table 21, below. Conservatively assuming that a TI-SGTR event is a Large Early Release event, the sum of each of the Table 21 columns is the total TI-SGTR LERF for the associated burnup.

Table 21.

TI-SGTR Vulnerability Category	Frequency (/rx-yr)	TI-SGTR Frequency 2P99 After Repairs (14.96 EFPD)	TI-SGTR Frequency 2P00 No Repairs (15.46 EFPD)	TI-SGTR Frequency 2R14 No Repairs (15.81 EFPD)	TI-SGTR Frequency 2P00 With Repairs (15.46 EFPD)	TI-SGTR Frequency 2R14 With Repairs (15.81 EFPD)
H/D/Hi	2.763E-06	1.095E-08	1.958E-08	3.396E-08	1.095E-08	1.699E-08
H/D/Hc	1.464E-07	6.068E-10	1.055E-09	1.797E-09	6.068E-10	9.229E-10
H/D/Li	2.494E-06	2.232E-06	2.337E-06	2.398E-06	2.232E-06	2.306E-06
H/D/Lc	1.321E-07	1.237E-07	1.259E-07	1.269E-07	1.237E-07	1.252E-07
M/D/Hi	3.662E-06	3.978E-09	8.500E-09	1.748E-08	3.978E-09	7.027E-09
M/D/Hc	1.941E-07	2.204E-10	4.578E-10	9.247E-10	2.204E-10	3.816E-10
M/D/Li	3.306E-06	8.108E-07	1.015E-06	1.234E-06	8.108E-07	9.534E-07
M/D/Lc	1.752E-07	4.491E-08	5.464E-08	6.530E-08	4.491E-08	5.177E-08
SUM	**1.287E-05**	**3.227E-06**	**3.562E-06**	**3.879E-06**	**3.227E-06**	**3.462E-06**

The table shows that the dominant contributor to the TI-SGTR LERF is the High RCS pressure/Dry SG/Low SG pressure/intact RCP seal loop TI-SGTR category. This is to be expected.

For consistency, the TI-SGTR risk results in Table 21 will be re-written using the format presented for the Pressure Induced SGTR risk results presented in Section 4.3.3. As stated in earlier discussions, since the TI-SGTR is a consequence of core damage, it contributes nothing to CDF. Table 22 summarizes the TI-SGTR contribution to LERF as a function of burnup during the second half of ANO-2 Cycle-14. The average TI-SGTR LERF for each of the burnup interval is presented in Table 23.

Table 22. Temperature-induced SGTR Instantaneous LERFs

ANO-2 Burnup	Instantaneous CDF TI SGTR (/rx-yr)
BOP	3.227E-06
MOP-NR	3.562E-06
EOP-NR	3.879E-06
MOP-WR	3.227E-06
EOP-WR	3.462E-06

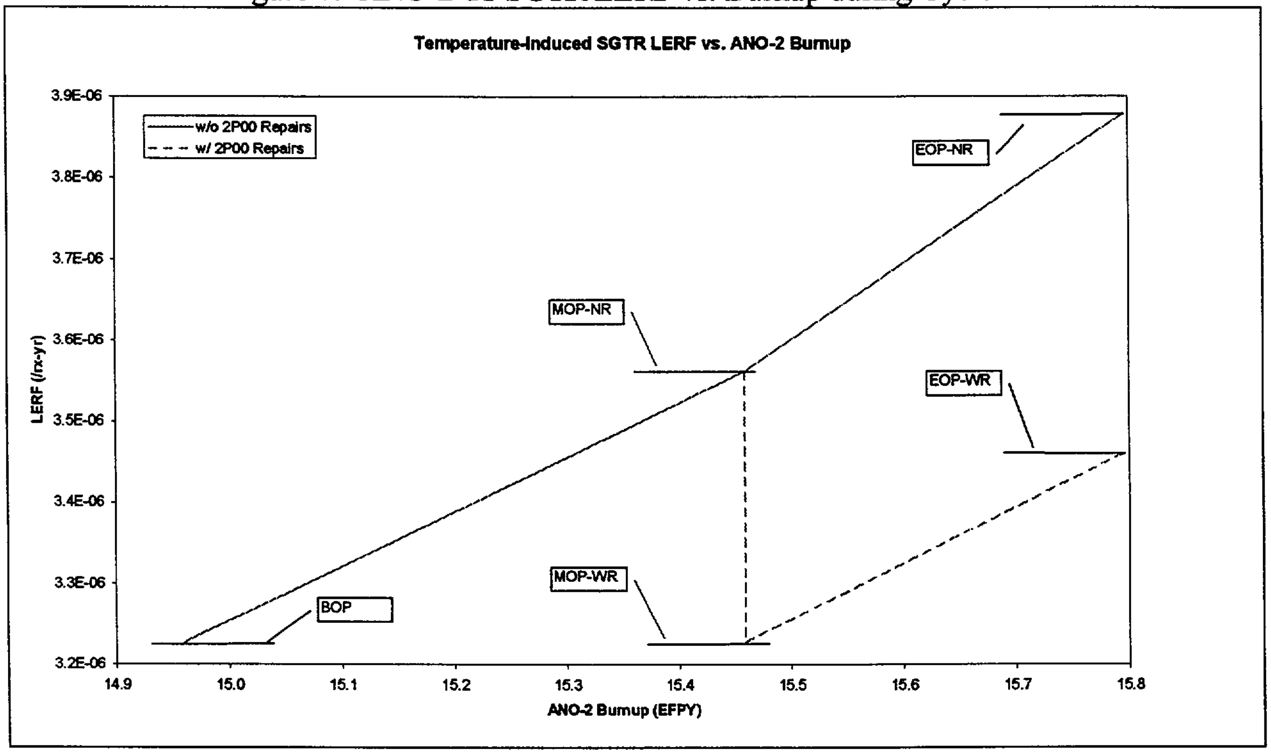
Table 23. Temperature-induced SGTR Average LERFs

ANO-2 Burnup Interval	Interval Average CDF TI SGTR (/rx-yr)
BOP to MOP-NR	3.395E-06
MOP-NR to EOP-NR	3.720E-06
MOP-WR to EOP-WR	3.345E-06

A plot of the TI-SGTR LERF results at BOP, MOP-NR, EOP-NR, MOP-WR, and EOP-WR is provided in Figure 7, below. As expected, the LERF increases over the period. At the proposed 2P00, assumed to begin on 5/15/00 at burnup 15.46 EFPY, the plot splits into two cases: (1) a

2P00 SG inspection/repair outage and (2) no 2P00 SG inspection/repair outage. For case (1), after the SG inspection/repair outage, the LERF is assumed to return to its value just after 2P99 and then to increase at about the same rate as it increased just after 2P99 until the end of the period at 2R14. For case (2), with no 2P00 SG inspection/repair outage, the LERF continues to increase monotonically beyond its value at MOP-NR. Its rate increases due to the increasing impact of the degradation process as predicted by the SG tube integrity data.

Figure 7. ANO-2 TI-SGTR LERF vs. Burnup during Cycle-14



4.5 Assessment of ANO-2 Risk Due to Spontaneous, Pi-SGTR, and TI-SGTRs

The overall ANO-2 CDF and LERF from all contributions is summarized in this section. The overall ANO-2 CDF includes the ANO-2 Baseline CDF plus the PI-SGTR CDF; the TI-SGTR contributes nothing to CDF. The Baseline CDF is 1.969E-5/rx-yr, as shown in Table 15. The instantaneous and average PI-SGTR CDF values are provided on Tables 5 and 6. The Baseline and PI-SGTR CDF values are summed on Tables 24 and 25.

Table 24. Total ANO-2 Instantaneous CDFs

ANO-2 Burnup	Instantaneous CDF Baseline + PI-SGTR + TI-SGTR (/rx-yr)
BOP	1.970E-05
MOP-NR	1.970E-05
EOP-NR	1.970E-05
MOP-WR	1.969E-05
EOP-WR	1.969E-05

Table 25. Total ANO-2 Average CDFs

ANO-2 Burnup Interval	Interval Average CDF Baseline + PI-SGTR + TI-SGTR (/rx-yr)
BOP to MOP-NR	1.970E-05
MOP-NR to EOP-NR	1.970E-05
MOP-WR to EOP-WR	1.969E-05

The overall ANO-2 LERF includes the ANO-2 Baseline LERF (which includes both Spontaneous SGTR and ISLOCA LERFs), the PI-SGTR LERF, and the TI-SGTR LERF. The Baseline LERF, calculated to be 1.572E-06/rx-yr in Section 4.4.2.1, is a constant with respect to burnup. As noted in Section 4.3, the PI-SGTR LERF is assumed to be equal to the PI-SGTR CDF. Thus, the instantaneous and average PI-SGTR LERF values are provided in Tables 5 and 6. The instantaneous and average TI-SGTR LERF values are provided in Tables 22 and 23. Summing these terms, the instantaneous and average LERFs are listed in Tables 26 and 27.

Table 26. Total ANO-2 Instantaneous LERFs

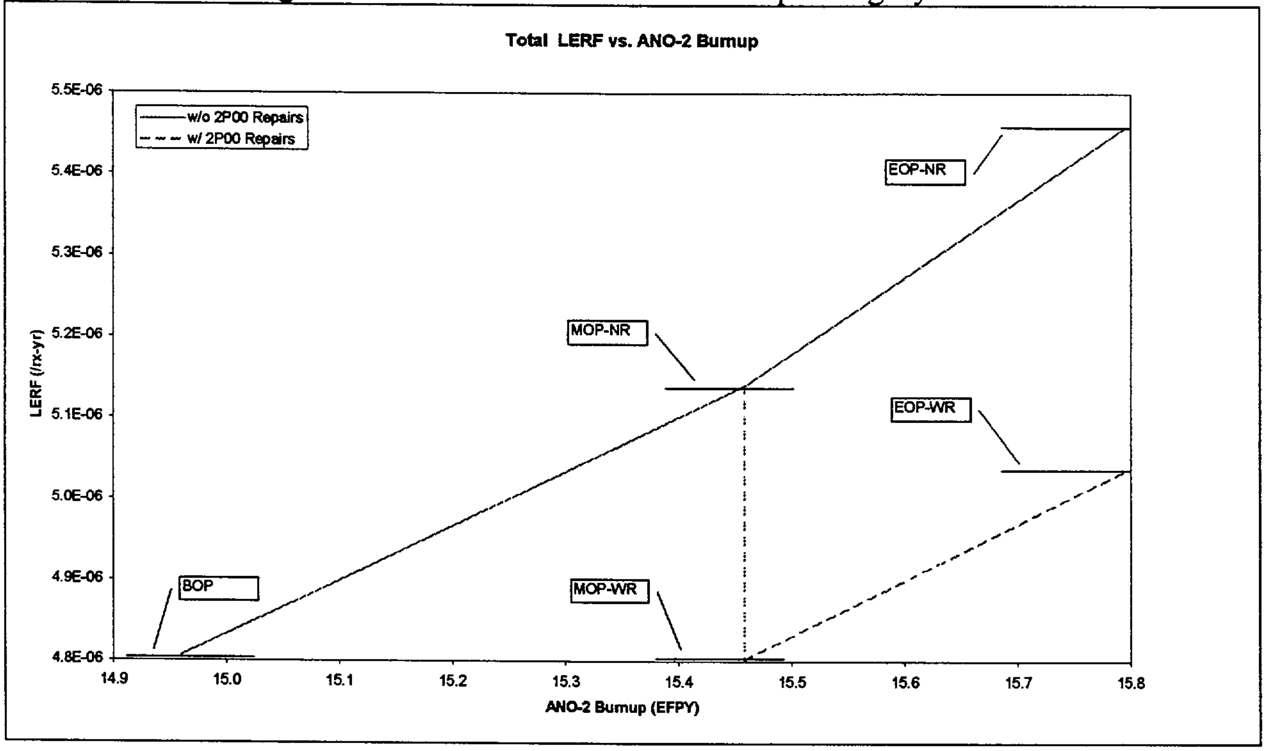
ANO-2 Burnup	Instantaneous LERF Baseline + PI-SGTR + TI-SGTR (/rx-yr)
BOP	4.806E-06
MOP-NR	5.140E-06
EOP-NR	5.458E-06
MOP-WR	4.802E-06
EOP-WR	5.037E-06

Table 27. Total ANO-2 Average LERFs

ANO-2 Burnup Interval	Interval Average LERF Baseline + PI-SGTR + TI-SGTR (/rx-yr)
BOP to MOP-NR	4.973E-06
MOP-NR to EOP-NR	5.299E-06
MOP-WR to EOP-WR	4.919E-06

Since the PI-SGTR CDF is a very small contributor to the total ANO-2 CDF, the total CDF is essentially constant over the second half of Cycle-14. No plot of CDF vs. burnup is necessary. However, the total ANO-2 LERF is burnup dependent. A plot of the total LERF is provided in Figure 8, below.

Figure 8. ANO-2 Total LERF vs. Burnup during Cycle-14



4.6 Comparison of ANO-2 Risk Results with Regulatory Guidelines

In this section, the CDF results reported in Tables 24 and 25 are used to generate PI- and TI-SGTR ΔCDF values and the LERF results reported in Tables 26 and 27 are used to generate PI- and TI-SGTR ΔLERF values. These ΔCDF and ΔLERF values are then compared with NRC risk acceptance guidelines on these parameters provided in NRC Regulatory Guide 1.174 [Ref. 2] to determine the risk impact of a SG Inspection/Repair outage during the second half of ANO-2 Cycle-14.

Regulatory Guide 1.174 [Ref. 2] provides the Staff's recommendations for using risk information in support of license basis changes requiring Staff review and approval. The guide is intended to provide consistency in regulatory decisions in areas in which the results of risk analyses are used to help justify regulatory action. The guidance establishes risk-acceptance guidelines. These guidelines utilize two risk metrics: CDF and LERF. For each, the magnitude of acceptable changes in the metric (i.e., ΔCDF and ΔLERF) are specified for a given baseline value of the metric (CDF and LERF). For each metric, three regions of acceptance are defined in Tables 28 and 29, as follows.

Table 28. Reg Guide 1.174 Acceptance Guidelines for CDF

Region	CDF	ΔCDF	Implication
I	$<10^{-4}$/rx-yr	$\geq10^{-5}$/rx-yr	No changes allowed
I	$\geq10^{-4}$/rx-yr	$\geq10^{-6}$/rx-yr	No changes allowed
II	$<10^{-4}$/rx-yr	$\geq10^{-6}$/rx-yr and $<10^{-5}$/rx-y	Small changes
III	all	$<10^{-6}$/rx-yr	Very small changes

Table 29. Reg Guide 1.174 Acceptance Guidelines for LERF

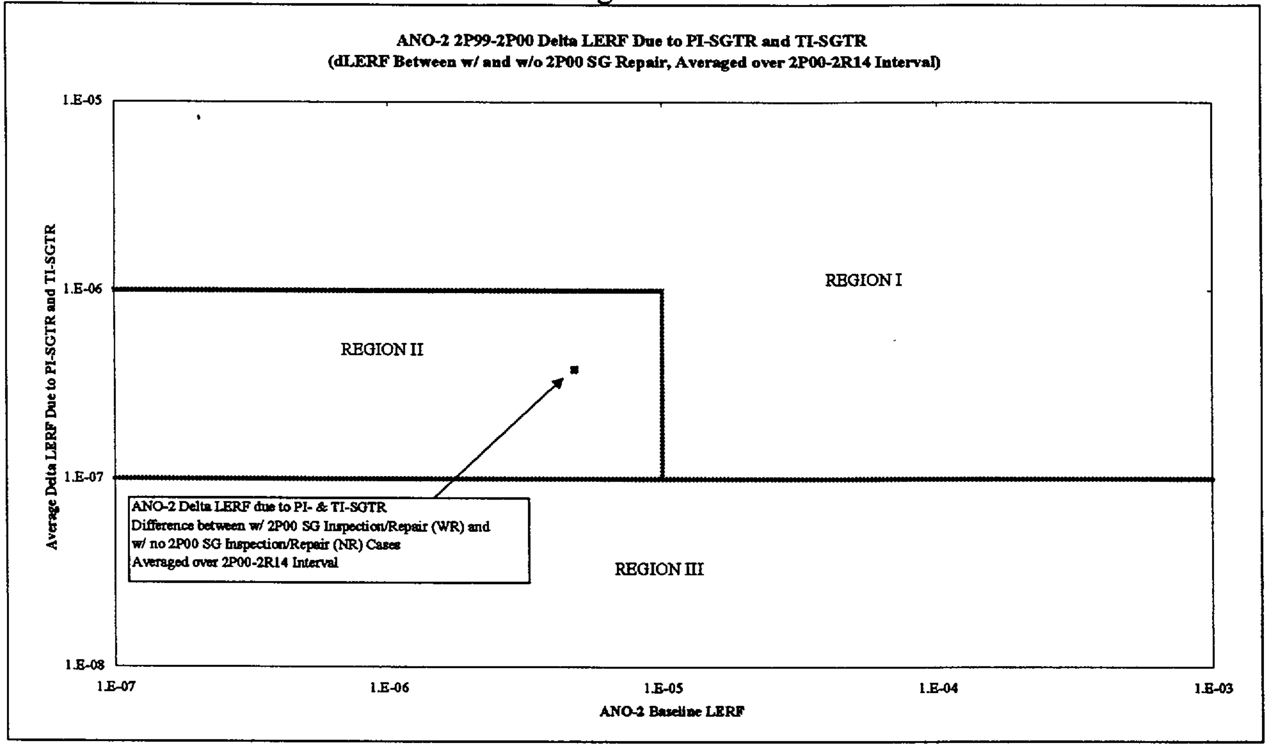
Region	LERF	ΔLERF	Implication
I	$<10^{-5}$/rx-yr	$\geq10^{-6}$/rx-yr	No changes allowed
I	$\geq10^{-5}$/rx-yr	$\geq10^{-7}$/rx-yr	No changes allowed
II	$<10^{-5}$/rx-yr	$\geq10^{-7}$/rx-yr and $<10^{-6}$/rx-y	Small changes
III	all	$<10^{-7}$/rx-yr	Very small changes

The Reg Guide, however, does not explicitly define either the baseline metric or the change in the metric. This definition will depend on the application. Several interpretations for these metrics are possible for the SGTR risk issue. A few of these will be explored to assess the acceptability of the planned ANO-2 SG operating strategy.

First, using the information in Tables 24 and 25, it is noted that by any measure, the ANO-2 CDF is well below 10^{-4}/rx-yr and the change in CDF associated with the SG inspection/repair outage is much less than 10^{-7}/rx-yr. Thus, from a CDF perspective, the SG inspection/repair outage represents a "very small change" and should be acceptable from a regulatory position.

The change in LERF between the 2P00 SG inspection/repair outage (With Repair, WR) option and the no SG inspection/repair outage (No Repair, NR) option is not so clear. The LERF and ΔLERF values associated with these options can be calculated in several ways. One way is to calculate the average ΔLERF between the NR and the WR cases for the MOP to EOP operating interval. Using this definition and the information in Tables 26 and 27, the average ΔLERF between the NR and WR cases in the MOP to EOP operating interval is 3.8E-07/rx-yr (i.e., 5.299E-06 - 4.919E-06). This definition will be called $\Delta LERF_1$ for later reference. Assuming that the baseline LERF is the total LERF at BOP, 4.806E-06/rx-yr, then per the LERF acceptance guidelines, the effect of the SG inspection/repair outage is small since it falls in Region II. A plot of ΔLERF vs. LERF is shown on Figure 9, below.

Figure 9.



This definition of ΔLERF does not reflect the risk difference between the two operating strategies, since it does not account for the length of time that the plant will experience an increased LERF. By way of example, using this definition, the ΔLERF associated with a SG inspection/repair shortly after 2P00 would be very small and one just before 2R14 would be very large, yet the average LERF for the two cases would be nearly identical. *Since risk is a stochastic parameter, risk measures should be time-weighted.*

Given that the issue of interest is the risk difference between the two operating strategies (NR or WR) and given that the SG Inspection/Repair could occur at any time during the second half of ANO-2 Cycle-14 until a design or licensing basis is exceeded, then the difference between the average LERF for the two cases over the entire period (2P99 to 2R14) is a better risk impact measure than the definition presented above.

Using the ANO-2 Cycle-14 burnups at 2P99 (14.96 EFPY), 2P00 (15.46 EFPY), and 2R14 (15.81 EFPY) and the average LERF values in Tables 26 and 27, the average LERF with no and with a SG inspection/repair outage 2P00 are,

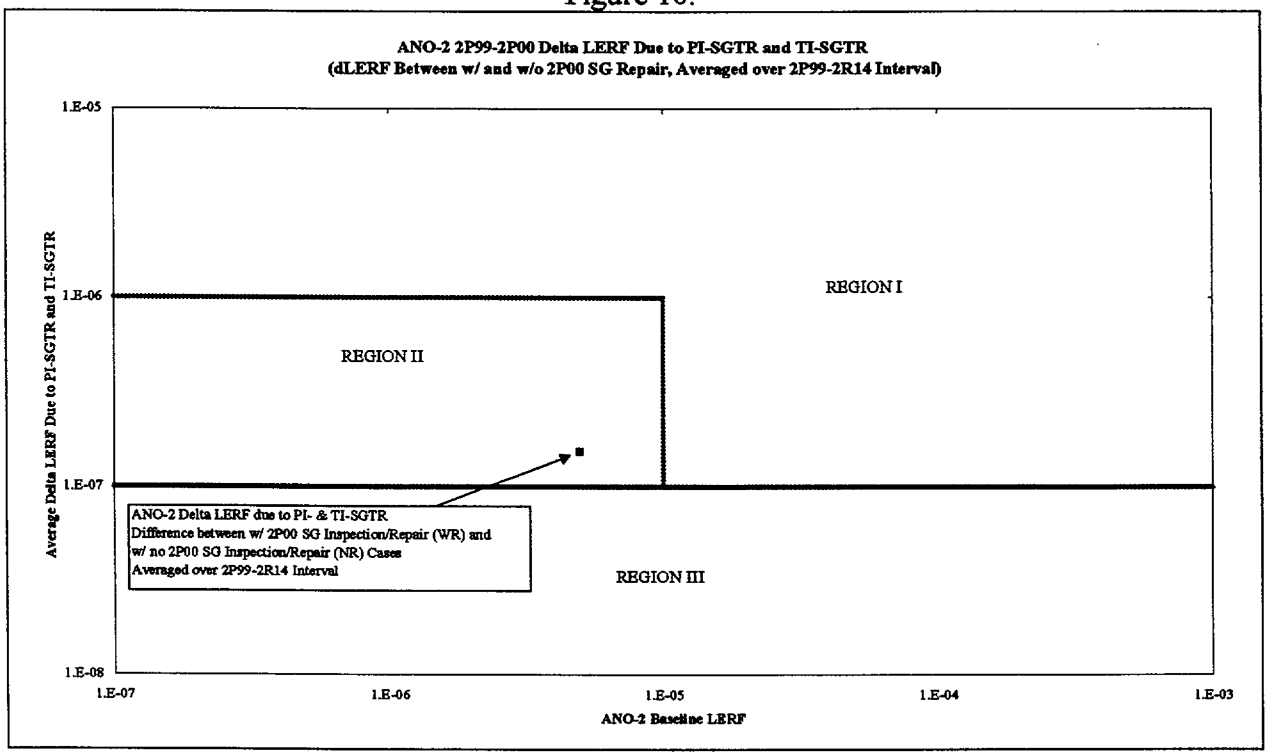
$$\text{Avg LERF(NR)} = \frac{(15.46-14.96)*(4.973\text{E}-06)+(15.81-15.46)(5.299\text{E}-06)}{(15.81-14.96)}$$
$$= 5.107\text{E}-06/\text{rx}-\text{yr}$$

and

$$\text{Avg LERF(WR)} = \frac{(15.46-14.96)*(4.973\text{E}-06)+(15.81-15.46)(4.919\text{E}-06)}{(15.81-14.96)} = 4.951\text{E}-06/\text{rx}-\text{yr}$$

The difference between these two values is a better definition for ΔLERF for the issue. Using the above results, ΔLERF equals 1.5E-07/rx-yr. This definition will be called $\Delta LERF_2$ for later reference. Again, assuming that the baseline LERF is the total LERF at BOP, 4.806E-06/rx-yr, then per the LERF acceptance guidelines, the effect of the SG inspection/repair outage is small since it falls in Region II. A plot of ΔLERF vs. LERF is shown on Figure 10, below.

Figure 10.



Another consideration in assessing the risk associated with the SG inspection/repair outage is the selection of the date of the proposed SG inspection/repair outage. The date proposed in this analysis is 5/15/00, which corresponds to 15.46 EFPY. This date was chosen for economic reasons (i.e., to avoid a summer outage). Without a licensing change, the inspection outage must start before the licensing basis or design basis is exceeded; however, operation could continue without a SG inspection/repair outage until the End of Design Margin (EDM). Based on conservative estimates of the EDM date developed shortly after 2P99, the EDM date occurs well after 5/15/00. Initial analyses estimated the EDM to occur about 6/30/00 (with a corresponding burnup of approximately 15.58 EFPY). Assuming that the SG outage starts at this burnup value and using the $\Delta LERF_2$ definition, the ΔLERF between the NR and WR cases is calculated as follows. The Average LERF(NR) is the same as above,

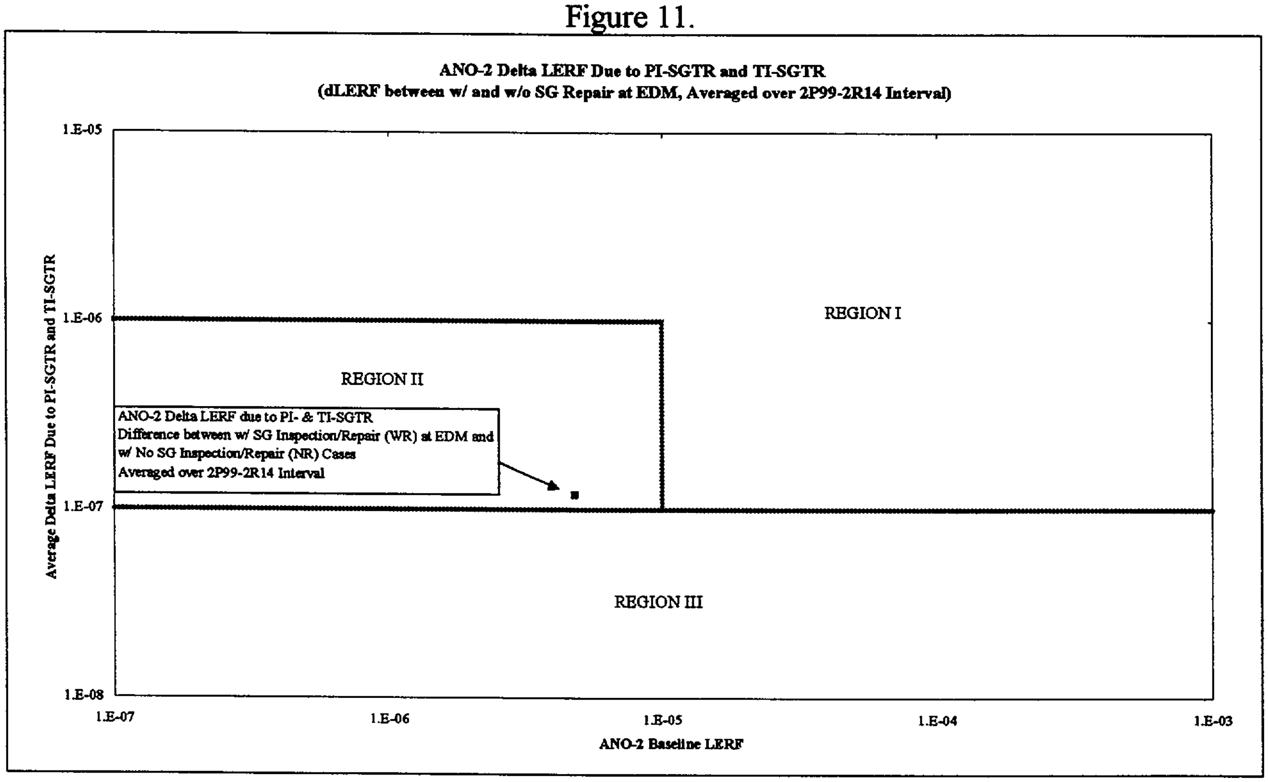
$$\text{Avg LERF(NR)} = \frac{(15.46-14.96)*(4.973\text{E}-06)+(15.81-15.46)(5.299\text{E}-06)}{(15.81-14.96)}$$
$$= 5.107\text{E}-06/\text{rx}-\text{yr}$$

The Average LERF(WR) can be estimated by interpolating the LERF values vs. burnup. Interpolating between the MOP-NR and EOP-NR LERF values in Table 26, the LERF at 15.58 EFPY is about 5.26E-06. Likewise, assuming that the LERF at MOP-WR equals the LERF at BOP and that the rate of increase in LERF after the SG inspection/repair outage is the same as that just after 2P99, the LERF at EOP-WR is about 4.89E-06. The average LERF(WR), thus, is the weighted sum,

$$\text{Avg LERF(WR)} = \frac{(15.58-14.96)(4.806\text{E}-6+5.26\text{E}-6)+(15.81-15.58)(4.806\text{E}-6+4.89\text{E}-6)}{(2)(15.81-14.96)}$$
$$= 4.983\text{E}-06/\text{rx}-\text{yr}$$

The difference between these terms yields the Average ΔLERF of 1.2E-07/rx-yr. This definition will be called $\Delta LERF_3$ for later reference. Again, assuming that the baseline LERF is the total LERF at BOP, 4.806E-06/rx-yr, then per the LERF acceptance guidelines, the effect of the SG inspection/repair outage is small since it falls in Region II. A plot of ΔLERF vs. LERF is shown on Figure 11, below.

Figure 11.

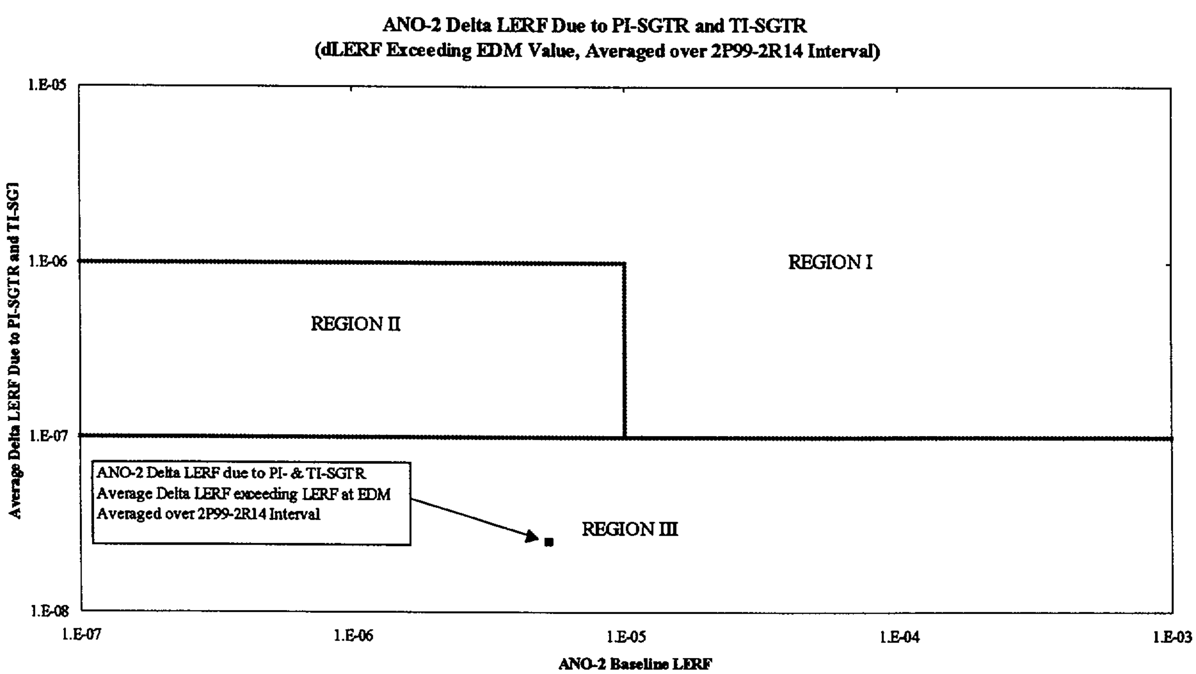


As already noted, if it is acceptable from a regulatory perspective to operate up to the "End of Design Margin (EDM)" (i.e., operating within the design margins has been, to date, considered "safe"), then it is appropriate to consider only risk increases beyond the design margin. From this perspective, regulatory concern should be limited to LERF increases beyond design margin. This interpretation leads to a fourth definition of ΔLERF: $\Delta LERF_4$ is one-half of the difference between the LERF at EOP-NR and the LERF at EDM averaged over the second half of Cycle-14. The LERF at EOP-NR is 5.458E-06 from Table 26. The LERF at EDM is 5.26E-06, from above. Thus, the $\Delta LERF_4$ is calculated as follows,

$$\Delta LERF_4 = \tfrac{1}{2}(5.458E\text{-}06 - 5.26E\text{-}06)(15.81-15.58)/(15.81-14.96)$$
$$= 2.5E\text{-}08/rx\text{-}yr$$

Again, assuming that the baseline LERF is the total LERF at BOP, 4.806E-06/rx-yr, then per the LERF acceptance guidelines, the effect of the SG inspection/repair outage is small since it falls in Region III. A plot of ΔLERF vs. LERF is shown on Figure 12, below.

Figure 12.



5.0 RESULTS/CONCLUSIONS/RECOMMENDATIONS

In summary, a Steam Generator Tube Rupture (SGTR) risk assessment performed in support of an ANO-2 Steam Generator (SG) Operational Assessment for the second half of ANO-2 Cycle-14 in response to Condition Report CR-ANO-2-1999-0727. These assessments estimate the risk associated with spontaneous Steam Generator Tube Ruptures (SGTRs), Pressure-Induced SG Tube Ruptures (PI-SGTRs), and Temperature-Induced SGTRs (TI-SGTRs) in a manner consistent with NUREG-1570 and the EPRI SG Tube Integrity Risk Assessment Methodology. The risk assessment assessed the risk benefits associated with a proposed SG Inspection/Repair outage (called 2P00) during the second half of Cycle-14. Both CDF and LERF values were generated to measure the risk impact of the proposed outage. The CDF and LERF results were compared with NRC risk acceptance guidelines for these parameters provided in NRC Regulatory Guide 1.174 [Ref. 2] to determine the acceptability of the option of operating from 2P99 to 2R14 without a SG inspection/repair outage.

The analysis showed that the SG inspection/repair outage had a very small risk impact (i.e., Region III) using Reg Guide 1.174 criteria. The LERF results indicated that the risk increase associated with operating from 2P99 to 2R14 without a SG inspection/repair outage was very small (i.e., Region II) to very small (i.e., Region III) depending on interpretation. Although the risk with continued operation is small, it will be further reduced, if not negated, by the risk associated with performing a SG inspection, an activity involving an extended period at reduced inventory. The averted risk was not quantified.

Awareness of the risk associated with a Temperature-Induced SG Tube Rupture has been raised at the plant and plant actions have been taken to minimize this risk. These actions include,

- Revision of plant Emergency Operating Procedures (EOPs) to assure that SGs remain pressurized should a loss of all feedwater occur,
- Revision of plant-specific Severe Accident Management Guidelines (SAMGs) to assure that the SGs remain pressurized should a loss of all feedwater occur,
- Revision of SAMGs to avoid the loss of Reactor Coolant Pump (RCP) loop seals should feedwater be lost to a SG associated with the RCP, and
- Most importantly, the current SGs will be replaced in upcoming 2R14 refueling outage.

Thus, the risk associated with operating ANO-2 from 2P99 to 2R14 without a SG inspection/repair outage is considered small and therefore acceptable from a risk perspective.

6.0 ATTACHMENTS

Attachment A	List of Attached Computer Files
Attachment B	Tube Burst Probability vs. Pressure Calculations
Attachment C	ANO-2 Rev-1 Accident Sequence Descriptions
Attachment D	Application of Human Reliability Assessment Methodology and Use of EOATS to Support ANO2 SG Tube Integrity Evaluations
Attachment E	Application Additional Operator Recoveries to ANO-2 Rev-1 Cutset Results
Attachment F	Arkansas Nuclear Unit One, Unit 2 External Events Investigation for Steam Generator Tube Integrity Risk Assessment
Attachment G	Arkansas Nuclear Unit One, Unit 2 Human Reliability Assessments Associated with Maintenance of SG Pressurization and RCP Loop Seals
Attachment H	MSSV Failure to Reclose Probability Estimates for SBO and Loss of DC Initiated Severe Accident Sequences at ANO-2

Attachment A
List of Attached Computer Files

The following files are provided as part of this calculation:

Disk 1:

Filename	Description	Date	Time
A2R1COM.ZIP	WINZip file	12/12/99	15:00

Contents of A2R1COM.ZIP:

Filename	Description	Date	Time
a2r1com.cut	Cutset file (modified version of @a2r1com.cut)	12/12/99	14:58
a2r1com.be	Basic Event file (modified version of a2r1com.be)	12/12/99	14:52
a2r1com.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00

Disk 2:

Filename	Description	Date	Time
A2R1COM1.ZIP	WINZip file	12/12/99	16:17

Contents of A2R1COM1.ZIP:

Filename	Description	Date	Time
a2r1com1.cut	Cutset file (modified version of @a2r1com.cut)	12/12/99	16:17
a2r1com1.be	Basic Event file (modified version of a2r1com.be)	12/12/99	16:05
a2r1com1.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com1.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00

Disk 3:

Filename	Description	Date	Time
A2R1COM2.ZIP	WINZip file	12/12/99	16:56

Contents of A2R1COM2.ZIP:

Filename	Description	Date	Time
a2r1com2.cut	Cutset file (modified version of @a2r1com.cut)	12/12/99	16:55
a2r1com2.be	Basic Event file (modified version of a2r1com.be)	12/12/99	16:53
a2r1com2.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com2.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00

Disk 4:

Filename	Description	Date	Time
A2R1COM3.ZIP	WINZip file	12/12/99	21:26

Contents of A2R1COM3.ZIP:

Filename	Description	Date	Time
a2r1com3.cut	Cutset file (modified version of @a2r1com.cut)	12/12/99	21:25
a2r1com3.be	Basic Event file (modified version of a2r1com.be)	12/12/99	16:53
a2r1com3.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com3.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00

Disk 5:

Filename	Description	Date	Time
A2R1COM4.ZIP	WINZip file	02/23/00	20:59

Contents of A2R1COM4.ZIP:

Filename	Description	Date	Time
a2r1com4.cut	Cutset file (modified version of @a2r1com.cut)	12/13/99	10:22
a2r1com4.be	Basic Event file (modified version of a2r1com.be)	12/13/99	09:36
a2r1com4.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com4.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00
BTDREC.txt	QRECOVER rule file for BTD	12/13/99	09:10
RUNREC.txt	QRECOVER rule file for RUN	12/13/99	09:15

Disk 6:

Filename	Description	Date	Time
A2R1COM5.ZIP	WINZip file	12/13/99	14:22

Contents of A2R1COM5.ZIP:

Filename	Description	Date	Time
a2r1com5.cut	Cutset file (modified version of @a2r1com.cut)	12/13/99	14:21
a2r1com5.be	Basic Event file (modified version of a2r1com.be)	12/13/99	14:19
a2r1com5.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com5.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00

Disk 7:

Filename	Description	Date	Time
A2R1COM6.ZIP	WINZip file	12/28/99	10:47

Contents of A2R1COM6.ZIP:

Filename	Description	Date	Time
a2r1com6.cut	Cutset file (modified version of @a2r1com.cut)	12/28/99	10:40
a2r1com6.be	Basic Event file (modified version of a2r1com.be)	12/13/99	14:19
a2r1com6.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com6.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00

Disk 8:

Filename	Description	Date	Time
A2R1COM7.ZIP	WINZip file	02/23/00	22:38

Contents of A2R1COM7.ZIP:

Filename	Description	Date	Time
a2r1com7.cut	Cutset file (modified version of @a2r1com.cut)	02/23/00	22:36
a2r1com7.be	Basic Event file (modified version of a2r1com.be)	01/10/00	15:22
a2r1com7.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com7.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00
T10RECS.txt	QRECOVER rule file for OA-T10-RUN and OA-T10-STR	01/07/00	08:42

Disk 9:

Filename	Description	Date	Time
A2R1COM8.ZIP	WINZip file	01/11/00	21:56

Contents of A2R1COM8.ZIP:

Filename	Description	Date	Time
a2r1com8.cut	Cutset file (modified version of @a2r1com.cut)	01/11/00	21:55
a2r1com8.be	Basic Event file (modified version of a2r1com.be)	01/11/00	15:22
a2r1com8.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com8.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00

Disk 10:

Filename	Description	Date	Time
A2R1COM9.ZIP	WINZip file	01/11/00	22:21

Contents of A2R1COM9.ZIP:

Filename	Description	Date	Time
a2r1com9.cut	Cutset file (modified version of @a2r1com.cut)	01/11/00	22:19
a2r1com9.be	Basic Event file (modified version of a2r1com.be)	01/11/00	22:18
a2r1com9.gt	Gate file (identical to a2r1com.gt)	06/05/97	10:31
a2r1com9.tc	Type Code file (identical to a2r1com.tc)	01/01/97	01:00
HRArules.txt	OA2 rule file	01/11/00	20:44

Disk 11:

Filename	Description	Date	Time
ADV.ZIP	WINZip file	02/24/00	11:08

Contents of ADV.ZIP:

Filename	Description	Date	Time
ANO2R1.caf	ANO-2 Rev-1 Fault Tree Model file	04/16/97	08:12
ANO2R1ps.be	Basic Event file (identical to ano2r1ps.be)	11/04/97	10:58
ANO2R1ps.gt	Gate file (modified version of ano2r1ps.gt)	06/05/97	10:31
ANO2R1ps.tc	Type Code file (identical to ano2r1ps.tc)	01/01/97	01:00
ADVS.caf	ADV FTO fault tree (built from ANO-2 Rev-1 model)	01/29/00	22:12
ADVS.cut	Cutset file resulting from quantification of ADVS.caf	01/29/00	22:15

A2R1c7-a.cut	Results of DELTERM of ADVs.cut from A2R1com7.cut	01/29/00	22:20

Disk 12:

Filename	**Description**	**Date**	**Time**
SPRDSHTS.ZIP	WINZip file	3/09/00	11:06
DOCS.ZIP	WINZip file	3/09/00	n/a

Contents of SPRDSHTS.ZIP:

Filename	**Description**	**Date**	**Time**
FLBSLB-2P99.xls	EXCEL Spreadsheet for FLB/SLB-Induced SGTR CDFs, ΔCDFs, LERFs, and ΔLERFs using Eggcrate POBs from 2P99 data	2/21/00	13:23
ATWS-2P99.xls	EXCEL Spreadsheet for ATWS-Induced SGTR CDFs, ΔCDFs, LERFs, and ΔLERFs using Eggcrate POBs from 2P99 data	2/21/00	12:55
HML-WD.xls	EXCEL Spreadsheet for sorting CDF according to RCS pressure (High, Medium, Low) and SG Inventory (Wet, Dry)	3/07/00	13:32
TISGTR-2P99..xls	EXCEL Spreadsheet for Temperature-Induced SGTR LERF and ΔLERFs using Eggcrate POBs from 2P99 data	3/07/00	13:32

Contents of DOCS.ZIP:

Filename	**Description**	**Date**	**Time**
99E001901R0Main.doc	WORD document of Main Body for this calculation	3/09/00	n/a
99E001901R0Atts.doc	WORD document of Attachments for this calculation	3/09/00	n/a

Attachment B
Steam Generator Tube Probability of Burst Calculations

B.1 Introduction

Best estimate Steam Generator (SG) tube probability of burst values were estimated as a function of the pressure difference between the primary and secondary regions and as a function of burnup for the most limiting Steam Generator during the second half of ANO-2 Cycle 14. The probability of burst (POB) values for eggcrate axial defects were based on data collected during the ANO-2 SG Inspection/Repair Outage 2P99.

B.2 Eggcrate Axial Defects at 2P99

The burst probability verses pressure for the eggcrate axials at 2P99 were determined by varying the steam line break pressure input into the OPCON model. The output from the model is used to generate a pressure to burst correlation at 14.96 EFPY (the Beginning of Period (BOP)), at 15.46 EFPY (called Middle of Period (MOP)), and at 15.81 EFPY (the End-of-Cycle 14 burnup, called End of Period (EOP)). In the OPCON model the burst correlation is based on a 95/95 upper limit, while the input into the severe accident analysis uses a best estimate value. Both were provided. To acquire the best estimate values the pressure to burst correlation generated by the OPCON model is adjusted by adding 1000 psi to the pressure. The POB results for eggcrate defects at MOP just after a proposed 2P00 SG inspection outage was assumed to be identical to that at BOP just after 2P99. The results of the calculations are presented in Table B-1 and Figure B-1.

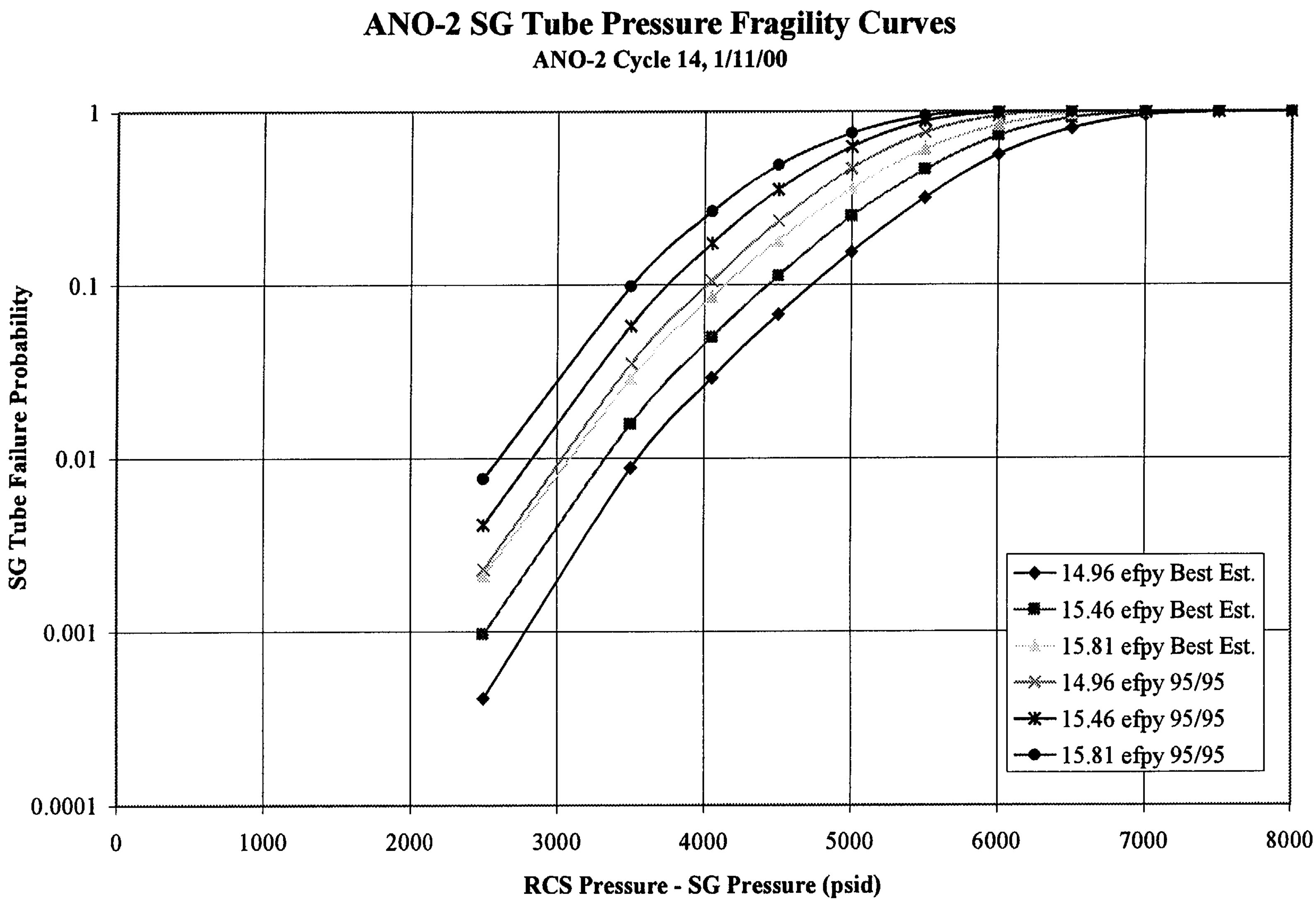
Table B-1 SG Tube Probability of Burst for EGGCRATE Defects Based on Data Collected at 2P99

	Probability of Burst										
RCS/SG Pressure Differential (psid)	**2500**	**3500**	**4050**	**4500**	**5000**	**5500**	**6000**	**6500**	**7000**	**7500**	**8000**
Eggcrate @BOP (14.96 EFPY) Best Estimate	0.000414	0.008803	0.029078	0.067469	0.155	0.319098	0.564983	0.8	0.955	0.995	1
Eggcrate @MOP (15.46 EFPY) Best Estimate	0.000971	0.015816	0.049952	0.112653	0.25	0.462326	0.726071	0.92	0.985	1	1
Eggcrate @EOP (15.81 EFPY) Best Estimate	0.002088	0.028751	0.085386	0.180925	0.36	0.604271	0.838664	0.97	1	1	1
Eggcrate @BOP (14.96 EFPY) 95/95	0.002274	0.03511	0.105795	0.233689	0.47	0.754842	0.946289	0.995	1	1	1
Eggcrate @MOP (15.46 EFPY) 95/95	0.004139	0.058	0.173506	0.355342	0.625	0.884035	0.985952	1	1	1	1
Eggcrate @EOP (15.81 EFPY) 95/95	0.007607	0.097923	0.265946	0.490869	0.75	0.946563	0.995806	1	1	1	1

Notes:

1. ANO-2 SG fragility data calculated by APTEC and supplied to ANO on 1/11/00. The fragility data is based on data from 2P99.
2. The 5000, 6500, 7000,7500, and 8000 psid data are estimated based on the following:
   a) 15.81 EFPY 95/95 is assumed to be 1 at 6500psid.
   b) The shape of the other curves are estimated based on shape of 15.81 EFPY 95/95 curve.
   c) The 5000 psid data is interpolated based on shape of each curve.

Figure B-1 SG Tube Probability of Burst for Eggcrate Defects Based on Data Collected at 2P99



Attachment C
ANO-2 Rev-1 Accident Sequence Descriptions

The following is a description of the ANO-2 Accident Sequences as reported in the ANO-2 PSA Rev-1 Summary Report [Ref. 7].

Transient-Initiated Core-Damage Sequences:

TQU/TQX - These sequences represent transient initiating events with successful reactor trip and successful primary-to-secondary heat transfer via the steam generators. However, subsequent failures induce a small break LOCA (such as PSVs failing to reclose or RCP seal LOCAs). This event is then treated by a transfer to the small break LOCA event tree (transfer S1) where potential subsequent failures of the HPSI system during injection from the RWT and recirculation from the containment sump are modeled. TQU sequences lead to early core melt, while TQX sequences lead to late core damage.

TBX - This sequence represents a transient initiating event followed by failure of RCS and core heat removal, but successful once-through-cooling. This results in the depletion of the RWT inventory and a requirement for recirculation of the containment sump inventory, which subsequently fails. This sequence represents a transient-induced medium LOCA. This event is treated by transfer S1 to the small break LOCA event tree. This sequence leads to late core damage.

TQBU - This sequence represents a transient initiating with subsequent failures resulting in a small break LOCA (such as PSVs failing to reclose or RCP seal LOCAs) with a subsequent failure of primary to secondary heat transfer via the steam generators and failure of HPSI to inject when aligning once through cooling.

TQBF - This sequence represents a transient initiating with subsequent failures resulting in a small break LOCA (such as PSVs failing to reclose or RCP seal LOCAs) with a subsequent failure of primary to secondary heat transfer via the steam generators and failure of once through cooling as a result of a failure to depressurize the RCS.

TQBX - This sequence represents a transient initiating with subsequent failures resulting in a small break LOCA (such as PSVs failing to reclose or RCP seal LOCAs) with a subsequent failure of primary to secondary heat transfer via the steam generators and failure of long term containment heat removal during once through cooling.

TBF - This sequence involves transient initiating events with a subsequent loss of RCS and core heat removal (i.e., MFW and EFW failures) and failure of once-through-cooling (due to HPSI or ECCS vent and LTOP vent valve failures). This sequence leads to high RCS pressure early core damage.

Small-Break LOCA Core-Damage Sequences:

SX - This sequence represents a small break LOCA with successful RCS inventory control and RCS and core heat removal, with a subsequent failure of HPSI or CS systems during recirculation (after the RWT inventory is exhausted). This leads to a late core damage.

SU - This sequence represents a small break LOCA with failure of the HPSI to replace inventory lost out the break. This event leads to an early core damage.

SBX - This sequence represents a small-break LOCA with successful HPSI, but with initial failure of RCS and core heat removal and recovery via once-through cooling. From this point on, this sequence is similar to SX above with a subsequent failure of the HPSI or CS systems during recirculation (after the RWT inventory is exhausted). This leads to a late core damage.

SBF - This sequence represents a small-break LOCA and failure of RCS and core heat removal through the steam generators and failure of once-through cooling. This leads to repressurization of the RCS above the HPSI shutoff head and early core damage at high RCS pressure.

SBU - This sequence represents a small-break LOCA with failure of the HPSI to replace inventory lost out the break. It also involves failure of the RCS and core heat removal function. Since this event is non-minimal when compared to sequence SU it is bounded by and considered within SU. However, for completeness, this sequence was included. This event leads to an early core damage.

<u>Medium-Break LOCA Core-Damage Sequence Summary</u>

MX - This sequence represents a medium-break LOCA with successful RCS inventory control and subsequent failure of HPSI or CS systems during recirculation (after the RWT inventory is exhausted). This leads to a late core damage.

MU - This sequence represents a medium-break LOCA with failure of the HPSI to replace inventory lost out the break. This event leads to an early core damage.

MK - This sequence represents a medium-break LOCA initiating event followed by failure of the reactor trip system. This sequence is not considered further due to the low frequency compared to transient-initiated ATWS events [Reference 13].

<u>Large-Break LOCA Core-Damage Sequences</u>

AX - This sequence represents a large-break LOCA with failure of the HPSI system or containment cooling function during recirculation (after the RWT is emptied). Due to the break size, the large flow rate required of low-pressure safety injection (LPSI), coupled with containment spray actuation, results in rapid depletion of the RWT inventory. Therefore, failure of recirculation following a large-break LOCA has been conservatively assumed to result in an early core damage.

AU - This sequence represents a large-break LOCA with failure of LPSI, HPSI or SITs. This sequence results in an early core damage and is assumed to occur very fast such that no operator recoveries are credited.

Steam Generator Tube Rupture Core-Damage Sequences

RX - This sequence represents the case when a SGTR occurs followed by successful reactor trip, primary-secondary heat removal, and inventory make-up. However, the RCS remains at high pressure and inventory is conservatively assumed to be lost through to SG. RCS inventory control will be lost when the RWT is depleted, and the RAS signal causes HPSI suction to be aligned to an empty containment sump. This sequence results in a late core damage.

RU - This sequence represents a SGTR followed by failure of the operators to use the unaffected steam generator to depressurize the RCS below the affected steam generator pressure (i.e., failure to terminate the leak) and failure of the HPSI system to make-up inventory lost out the ruptured tube. Although this sequence is slow to progress it has been conservatively assumed that this sequence results in early core damage.

RBX - This sequence represents the case where the SGTR occurs followed by loss of RCS and core heat removal. This requires once-through cooling, and requires that all decay heat be removed via the containment spray system during recirculation. If the flow path from the affected Steam generator is not isolated prior to the RWT inventory being exhausted from the sump, inventory control will be lost, leading to eventual core uncovering and late core damage.

RBF - This sequence represents a SGTR followed by success of RCS inventory control and loss of RCS and core-heat removal function. This results in RCS pressurization above the HPSI shutoff head. When once-through cooling is not successfully initiated, boiloff of RCS inventory out the SRVs results in a high RCS pressure early core damage.

RBU - This sequence represents the case where the SGTR occurs followed by loss of RCS and core heat removal. This requires once-through cooling, which subsequently fails. This sequence results in high RCS pressure early core damage.

RK - This sequence represents a SGTR followed by failure of the reactivity control function. This sequence is not considered further due to the low frequency compared with transient-initiated ATWS events [Reference 13].

ATWS Core-Damage Sequences

TK - This sequence represents transient initiating events followed by failure of the reactor trip system. These sequences have been considered separately in the ATWS work package [Reference 13].

SK - This sequence represents a small break LOCA initiating event followed by failure of the reactor trip system. This sequence is not considered further due to the low frequency compared to transient-initiated ATWS events [Reference 13].

Attachment D
Application of Human Reliability Assessment Methodology and Use of EOATS to Support ANO-2 SG Tube Integrity Evaluations

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D. Assessment of Recovery Actions for Use in Evaluating SG Tube Integrity at ANO-2

D.1 Defining the Accident Context

The key internal risk sequence in the PRA Model Rev1 is initiated by loss of a DC safety Bus (initiators T10 and T11). For these initiators, the loss of a DC bus is assumed to result in a turbine trip with the loss of main feedwater and closure of the MSIVs. This sequence also results in failure of the associated AC power buses to transfer to offsite power sources, and EDGs to start. Thus, this group of initiators was modeled to result in the loss of one half of the DC and AC buses. In this case one train of the EFW system (either the motor-driven or the turbine-driven) provides feed to the SGs, if not failed. The highest frequency core damage sequences include an independent failure of the remaining EFW train. If no operator actions are taken to restore equipment, this group of sequences is assumed to result in core damage with a high primary pressure, a dry SG, and a low pressure in the steam generator due to failure of a MSSV to reclose.

The basic elements of this sequence apply to all sequences involving loss of the secondary heat source for an extended period. Some proceduralized recovery paths may be limited by the context of the accident sequence.

D.2 HRA Issues (procedures, training, PSF's)

Application of the EOPs in response to a T10 initiator provides the operators with guidance and priorities for establishing one of five cooling success paths which will prevent core damage by maintaining sufficient cooling for the decay heat levels in the core. Success paths 1 and 3, which use electrically driven pumps (EFW and AFW), depend on AC and DC power supplies, which can be operated at manual locations within the plant. Success path 2 using the turbine driven pump is powered by steam from either steam generator; controls are supported by DC power if available, but can be operated without DC power locally. Success path 4 requires DC, AC and operator actions to reset protective trips and to line up the MFW/Condensate pathway. Success path 5, RCS depressurization, requires AC power for HPSI injection, and at least one division of DC power to open the high point vent pathway into the containment. The electrical system includes three large diesels available to Unit 2 (two safety and one auxiliary) and diverse offsite power sources that power the main and startup transformers. Connection of a source of power to the supply bus from any source of AC power can be accomplished locally by manually aligning breakers. The batteries provide more than eight hours supply to the DC safety buses.

If the actions are applied as indicated in the EOPs, success of any pathway will avoid core damage. Thus, the operators have significant time to recover feedwater in scenarios dominated by run failures, and once through cooling options if there is not enough time to recover the feedwater systems. The initial PRA is conservative in that it only uses one local recovery in a cutset, whereas the procedures specify local manual operations for recovery of any one of five diverse cooling pathways.

If a path becomes successful after core damage, but before tube damage this switches the high primary pressure/ low secondary pressure to a medium primary / high secondary pressure eliminating the thermal challenge condition. The severe accident management guidelines (SAMGs) support continued efforts to restore secondary cooling. The unique features of the SAMGs that change the pressure differential across the tubes are to:

- Initiate RCS depressurization to the containment in Candidate High Level Action 2 (CHLA-2) (this reduces creep rupture potential), and
- Depressurize the Steam Generators to enable injection (this increases creep rupture potential, and
- Restart RCS pumps to sweep non-condensable gases and clear loop seals to promote natural circulation (this increases creep rupture potential).
- There are notes that discuss the potential for isolating the steam generators, but no defined CHLA is listed (isolation of SGs when primary temperatures are high adds the secondary side as a barrier to release).

The key steps in evaluating the impact of operator actions on the potential for high/dry/low conditions was accomplished by first reviewing the ANO2 PRA study for sequences that threaten SG tube integrity under severe accidents. Then for representative sequences manual actions were identified that can be performed locally to maintain cooling and avoid SGTI challenges. To verify the ability to perform actions with EOP talk through and simulator observation of the T10 initiator verified steps used to manage high-dry sequence. Also the use of SAMGs was examined using a talk through method for key risk sequences. The approaches for integrating the HRA results into calculations of LERF and delta LERF using the current study structure were decided. Thus, the main EOAT was developed for incorporation of multiple recoveries with dependencies considered when assessing changes in LERF and delta LERF for next ISI cycle.

The risk quantification should address both internal and external events. The main difference between this sequence and the key ones that bound many external events is that the HSPI and some other AC equipment is available. Protection against generic external events can be evaluated using station blackout events as a surrogate sequence for all external events.

Walkthrough items

- Emergency lighting was available at each local site where a local recovery or repair action was postulated.

- All DC and AC breakers for aligning either offsite power, DG1, DG2 or the Station blackout DG to the 4160 volt bus were easy to get to, well labeled and color coded for the green or red division. Crosstie arrangements between buses require a special key to override this action which purposefully overrides the intended separation.

- Local breaker operational procedures and tools for operating the breakers were available in cabinets near each breaker location.

- There is no easy electrical or piping crosstie between units one and two. This reduces the chance of common cause failures between the units.

- Bus volt and amp meters are available on the breaker cabinets to note condition of the buses in addition to the breaker open-closed condition.

- Local feedback on the cause(s) of the station blackout DG failure to start or run through a computer system significantly cut the diagnosis time to nearly zero.

- Battery resource capability is very high (greater than eight hours) and loads such as flashing the DG fields are provided by local batteries.

- Local manual valve operations for opening or closing and controlling could be easily handled for EFW and AFW valves.

- Local Feedback on SG level is available at the feedwater inlet valves for the EFW and AFW inlets.

- The EFW turbine driven pump is located in a flood/fire/earthquake-protected environment. Local procedures are on the wall for repair of over-speed and other protective trips.

- All local control valves, breakers, and instrumentation were within the main plant buildings expect for the special station blackout DG which was in its own building.

- ANO2 uses CE symptom based emergency operating procedures, and functional recovery procedures. The procedures stress the recovery of the high-pressure systems for SG feed in the T10 sequence. Three diverse systems are provided for high pressure feed into the SGs.

D.3 Procedure review and Training Simulator

EOPs

Operators are trained on a full scope control room simulator. In a simulation of the loss of DC bus the crew pursued multiple paths for restoring one of three emergency/auxiliary feedwater systems: (1) the turbine driven emergency feedwater system, (2) the motor driven emergency feedwater system, and (3) the auxiliary motor driven feedwater system. (This permits use of systems designed for heat removal at plant nominal conditions.) A fourth option, which is cued when the steam generator level reached 70", is to open the ECCS Vent Valve, the Low Temperature Over Pressure (LTOP) valves, or the Pressurizer High Point Vent (HPV) valves connecting the pressurizer to the containment atmosphere. Opening any of these paths reduces primary pressure; opening the ECCS Vent Valve or an LTOP valve will reduce RCS pressure to the HPSI set point, which will allow HPSI flow to enter the RCS. In the simulation, the crew implemented this option as an anticipatory action when the level was about 90". This occurred at about 50 min +/- 20 min.; (some time was taken for discussion). This path was selected over the use of the condensate pumps because of HPSI availability, time to undue system isolations that were triggered during the event, potential for tripping the existing power circuits, and reports that none of the other systems could be repaired. The SG pressure stayed at about 600 psi, because the MSIVs closed due to protective signals, the turbine feed line was open and other small pipes remained open.

The simulator fidelity appeared to be very good. During the simulation an item noted is that one gauge for main feedwater flow continued to register when the flow was zero. A surprise was that the RCP trips on the lost bus control division did not stop the primary coolant pumps. Manually opening the LTOPs was accomplished by going to the back panels and using a key.

SAMGs

The SAMGs provide some cautions with regard to creep rupture, but it is not clear if this means the SG vessel or tubes. The primary objectives in the BD case are to get high pressure feedwater, then lower the pressure and carefully feed lower pressure feedwater to protect against tube damage. The steps for depressurizing the SGs to low pressure sources (condensate pressure) are identified, but not recommended unless the condensate is available.

There are some cautions, which could be interpreted as suggesting SG isolation to avoid creep rupture, but the step is not explicit and there is no cue for the action.

Linking the procedures to the PRA

Table 1 provides an assignment of the specific operator recovery actions called for in the procedures to a specific success path for preventing core damage in the T10 sequences. The functional objectives are represented in the EOATs as OARs. Within each OAR the recovery actions are treated as either parallel with dependencies or serial in achieving the success path. Additional dependencies are assessed between the tasks.

Table 1 Major functional events for SG tube integrity protection

Procedure	Major functional events	Operator Actions	Individual in CR	Individual Local	Expected time to complete local actions(mins)
EOPs	Reactor Trip	Manual Reactor Trip	CBOR		0
EOPs	Reactor Trip	Trip turbine	CBOR		0
EOPs	Reactor Trip	Trip RCPs	CBOR		0
EOPs	Coordinate shutdown EOPs	Prioritize Shutdown cooling options & Establish Emergency feedwater	CRS		10
Success 1	OAR-1				
EOPs	Establish hot shdwn cooling Electric EFW pump on loss of feed	Start DG A or B	CBOT or CBOR	OP1	10
EOPs	Establish hot shdwn cooling Electric EFW pump on loss of feed	Close DG breaker /Load train A 0r B	CBOT or CBOR	OP1	12
EOPs	Establish hot shdwn cooling Electric EFW pump on loss of feed	Start Electric EFW pump	CBOT or CBOR	OP2	13
EOPs	Establish hot shdwn cooling Electric EFW pump on loss of feed	Establish /control flow	CBOT or CBOR	OP2	15
EOPs	Establish hot shdwn cooling Electric EFW pump on loss of feed	Maintain SG level control	CBOT or CBOR	OP2	20
Success 2	OAR-2				
EOPs	Establish hot shdwn cooling Turbine Driven EFW pump on loss of feed	Establish Steam supply to TDEFWP	CBOT or CBOR	OP3	10
EOPs	Establish hot shdwn cooling Turbine Driven EFW pump on loss of feed	Establish suction water supply to TDEFWP	CBOT or CBOR	OP3	12
EOPs	Establish hot shdwn cooling Turbine Driven EFW pump on loss of feed	Establish Emergency feedwater to SG	CBOT or CBOR	OP3	13
EOPs	Establish hot shdwn cooling Turbine Driven EFW pump on loss of feed	Maintain SG level control	CBOT or CBOR	OP3	15
Success 3	OAR-3				
EOPs	Establish hot shdwn cooling AFW train on loss of feed	Start DG AA	CBOT or CBOR	OP1	20
EOPs	Establish hot shdwn cooling AFW train on loss of feed	Align breakers	CBOT or CBOR	OP1	25
EOPs	Establish hot shdwn cooling AFW train on loss of feed	Start Pump	CBOT or CBOR	OP2	28
EOPs	Establish hot shdwn cooling AFW train on loss of feed	Establish Emergency feedwater	CBOT or CBOR	OP2	29
EOPs	Establish hot shdwn cooling AFW train on loss of feed	Establish SG level	CBOT or CBOR	OP2	30
Success 4	OARCN-4				
EOPs	Inject into SGs before dry	Repair EFW, AFW	CRS	OP3	60
EOPs	Depressurize SG level before dry	Open ADVs	CBOT or CBOR	OP1 orOP2	60
Success 5	OARDC-5				
EOPs	Depressurize RCS before SG level is 70"	Verify HPSI is available	CRS		15
EOPs	Depressurize RCS before SG level is 70"	Open ECCS vent valves, LTOP valves, or RPV head vent valves	CRS		70
EOPs	Inject into the RCS before SG level is 70"	Insure that HPSI is flowing	CRS		72

D.4 HRA modeling

To develop the delta LERF calculation for accidents that challenge the SG tubes the HRA modeling assumptions were evaluated. The base PRA integrates recovery actions (restoring the function represented by the failed component) on a cutset by cutset basis. Application of the operator actions is done by initially assuming that the operator actions fail with a probability of one.

The recoveries are applied using "rule files" to automate the process. The rule files apply a specific recovery to a specific element or combination of elements in the cutset, thereby reducing the likelihood of the cutset by the non-recovery probability for the operator action. The base model assumes only one local (outside the control room) recovery in a cutset as verified by a cutset review. Multiple operator actions are allowed if one action is a standard control room action in the EOPs, and the other is a local start or control action. No equipment repair actions are modeled. The priority of selecting the recovery actions are (1) electrical and service water realignments, (2) recovery of the feedwater function, (3) then the recovery of other functions.

A list of the currently calculated recovery actions related to maintaining the SG tube boundary and providing a backup to the boundary are shown in Table 2. These elements are used to construct a multiple recovery EOAT tree assuming the T10 scenario.

Table 2 Operator actions representing multiple recoveries for success paths for T10

Name	HEP	Description	Type	Use in Success path
MANOSPREBD	2.80E-03	Operator fails to align offsite power to 2A1/2A2 after failed post trip auto-realign.	3	1
DGCRANK	3.11E-01	Operator fails to locally start DG followingfailure of start system (air or DC).	5	1
P7BMANREC	8.36E-02	Operator fails to manually control EFW pump (2P7B)discharge valves.	5	1
EFWXTIE	2.04E-01	Operator fails to open 2EFW-11A or 2EFW-11B and2CV-1025-1 or 2CV-1075-1, res	5	1
REC2D21&23	1.00E-01	Isolation of 2D21/2D23 from 2D01 and use of swing charger to power 2D21 and 2D	3	1
QHF2REFILL	1.25E-02	Operator fails to align EFW suction to alternate condensate source. (modeled)	5	1
QCSTKXFER	4.56E-02	Operator fails to align EFW suction to alternate ANO-1 condensate source.	5	2
P7AMANRECD	6.74E-02	Operator fails to manually control EFW pump (2P7A)speed and discharge valves; af	5	2
EDGAACRECR	1.23E-01	Operator fails to energize bus 2A3/2A4 from from AAC EDG (30 minute sequences	5	3
AFWFEEDREC	1.04E-02	Operator fails to start and align the AFW pump after loss of both EFW trains.	5	3
MANOSPRECD	5.69E-02	Operator fails to align offsite power to 2A1/2A2 after failed post trip auto-realign; for	5	4
ALTFEEDREC	3.79E-01	Operator fails to establish MFW or Cond to SG.	5	4
HOTLEGINJ	4.51E-02	Operator fails to align HPSI to Hot Leg Injectionfor R sequences with B success.	5	5
OPER-5A	4.38E-03	Operator fails to depressurize RCS using SCBCS.(modeled)	5	5

The HEPs for non-recovery are based on the TRC system, which assigns an error mode category, location, response time, time available, error factors, uncertainty factors. Defaults are provided based on the event categorization, and rules of thumb are provided for the application. This system is useful for single recovery models.

If multiple recoveries are to be used, dependencies must be addressed on a case by case basis. The means for considering a wider range of PSFs also needs to be established for this assessment. The key error modes recommended by the NRC should address detection, situation assessment, planning, and execution of the task (in the CR or locally). To provide an initial generic model for use with the cutsets a simple screening process is set up which assumes that one of the five paths is viable, and dependencies dominate the results. Based on the plant visit the second pathway could be classified as medium dependency (MD).

NUREG-1278 as shown in Table 3 provides factors for failure dependency between tasks. The categories are selected on the basis of timing, crew size, and shared equipment that has to be reset when switching from one success path to another. To start the assessment of adding multiple recoveries without a detailed assessment, Table 4 provides sufficient information to model secondary recovery modes.

Table 3 Inter-task dependency classifications

ZD	LD	MD	HD	CD
0.0	0.050	0.143	0.500	1.000

Table 4 Screening assessment non-recovery probabilities for multiple recoveries

	With AC	Without AC
First recovery	Detailed evaluation (0.05 if unknown)	Detailed evaluation (0.15 if unknown)
Second recovery	0.15	0.5
Third recovery	0.5	1
Fourth	1	1

This simple model permits an initial assessment of the recoveries to exercise the PRA model.

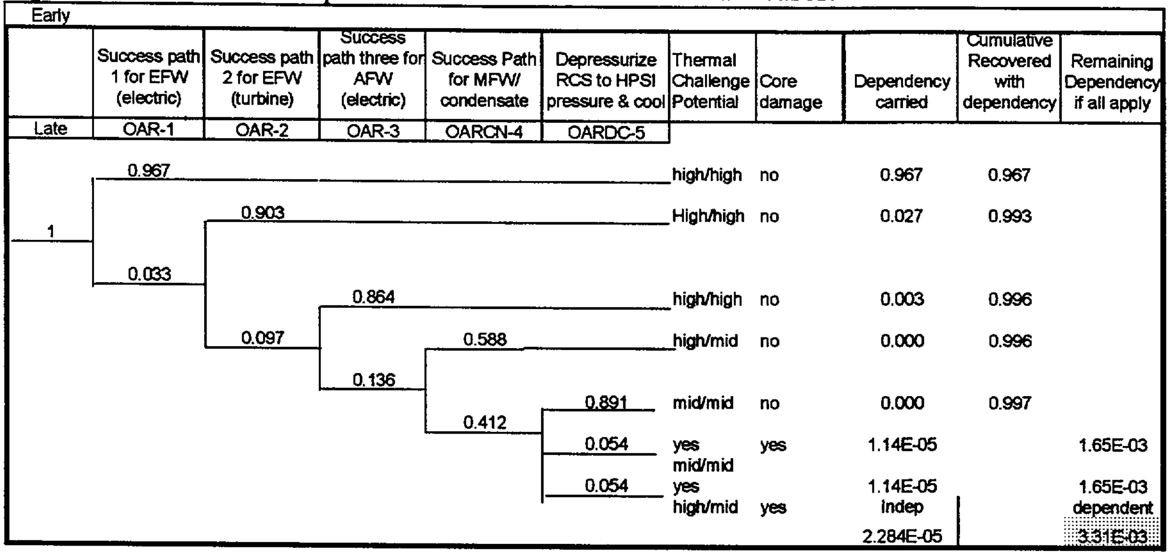
EOATs for quantification of integrated recovery actions using the EOPs

Based on the visit to the plant, review of the EOPs and SAMGs, and an accident simulation, it is very clear that the operational staff will pursue multiple recovery options at the same time. Thus, the structure of the HRA modeling for the accidents that challenge SG tube integrity can be improved by considering the multiple success paths for the four success cases shown in Table 1. The fifth success path tends to extend the time available for recovery of the other success paths, but would not reach 24 hours on its own.

A detailed EOAT model was developed to use the recoveries already shown in Table 2 to represent multiple recoveries for the T10 sequences that have AC power available. These include all feedwater recovery actions, and SG depressurization. One new recovery was added for the T10 sequences under OAR-1 (REC2D21&23 - Isolation of 2D21/2D23 from 2D01 and use of swing charger to power 2D21 and 2D23). This corrects DG failure to start and supply the motor driven EFW pump due to loss of control signals. The assessment was 0.1 with little or no time pressure and 0.15 in the early sequences with greater time pressure.

The following EOATs in Figures 1 and 2 provide an assessment for early core damage sequences that are initiated by T10 events. Each OAR is based on HRA database assessments that are expected to apply to the additional recoveries. OAR-1 includes recoveries from electrical faults that have a high dependence with OAR-3 and OAR-4. The use of the turbine driven motor is independent of electric power for pumping feedwater into the steam generators therefore low dependence was assigned. Medium dependence was assigned for OARCN-4 and OARDC-5 because these actions are planned for in the control room when the other success paths are not recovered within 15 to 20 minutes of the trip.

Figure 1 EOAT for multiple recoveries with failure to start cases.



The dependent result of 3.31E-3 for the early case is about two orders of magnitude greater than if all the actions were assumed to be independent. This value can be applied to the T10 sequences with start failures by substituting the non-recovery probability for all ex-control room actions for restoring feedwater and providing core cooling modeled in the cutsets. The routine EOP actions like manually tripping the reactor, or starting manual components that are called for in the procedures can remain in the cutsets (e. g. OPER actions), if the OPER dependency with these actions are included.

Table 5 Inputs to the early EOAT tree for multiple recoveries

Database Inputs from HEPs (Time adjusted HRA data base)					
Time adj	3.31E-02	9.69E-02	1.36E-01	4.12E-01	0.109
		MD	HD	MD	MD
Approximate Dependency Frac	OAR-1	OAR-2	OAR-3	OARCN-4	OARDC-5
OAR-1	1	0.112	0.043	0.018	0.008
OAR-2		1.000	0.062	0.054	0.024
OAR-3			1.000	0.075	0.034
OARCN-4				1.000	0.103
OARDC-5					1.000
Failure Dependency	0.033	0.112	0.104	0.148	0.170

Figure 2 repeats the same assessment for the late cases where T10 run failures are the main contributors that lead to a much longer time frame for recovery actions to be completed. In this case the result is 5.2 E-4 which is almost three orders of magnitude greater than an independent evaluation. The same rules apply as in the previous case.

Figure 2 EOAT for multiple recoveries with low time constraint

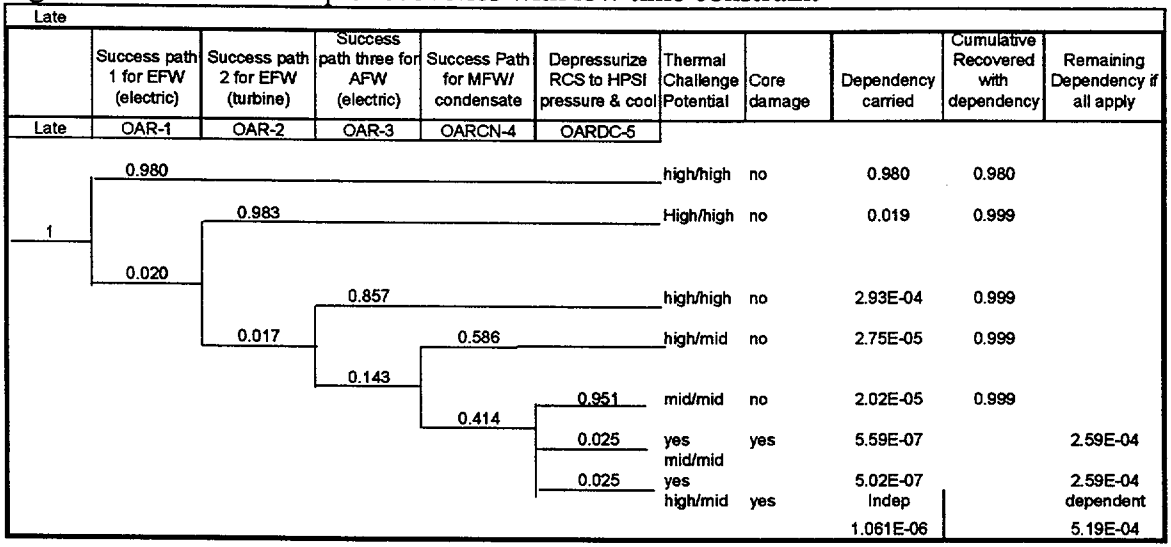


Table 6 summary of inputs for EOAT of Figure 2

Database Inputs from HEPs (Current HRA data base)					
Orig	2.00E-02	1.71E-02	1.43E-01	4.14E-01	0.049
		MD	HD	MD	MD
Approximate Dependency	OAR-1	OAR-2	OAR-3	OARCN-4	OARDC-5
OAR-1	1	0.026	0.019	0.011	0.004
OAR-2		1.000	0.011	0.010	0.003
OAR-3			1.000	0.080	0.027
OARCN-4				1.000	0.080
OARDC-5					1.000
Failure Dependency	0.020	0.026	0.030	0.100	0.114

Qualitative assessment of the SAMGs

The SAMGs list actions that can be used to mitigate the effects of a core damage event. For this assessment we assume that the core becomes uncovered. The evaluation examines the impact of applying the SAMG steps on plant states relative to The CHLAs for BD/B that affect SG tube integrity and barrier recovery are also listed in Table 7. Note that some of the actions are not clearly recommended in the SAMGs, so that the likelihood of completing them is very low in the current procedure form. The actions with a potential to increase thermal challenge to the SG tubes are CHLA-3 and 4. These actions are very good under some conditions, but if performed at the wrong time can increase the potential for loss of the tube barrier. A backup action for Isolating the SGs is proposed to reduce the potential for a LERF accident.

Table 7 Summary of SAMG actions that affect the steam generator tube integrity

Procedure for accident Mitigation	Major functional events	Cue for action	Operator Actions	Qualitative Thermal challenge impact	Expected time to complete local actions given early
BD SAMG CHLA-1	Inject into RCS replenish primary inventory	Core damage with depleted water inventory	Use available water sources for HPSI, LPI, or normal makeup pumps	Lowers pressure	>180
BD SAMG CHLA-7 (vent is –5)	Depressurize RCS to minimize prim. pressure and use film cooling	Core damage with water inventory in the vessel or low pressure supply available	High point vents with either train of DC available if not done in EOPs	Lowers pressure	>180
BD SAMG CHLA-2	Inject into SGs when dry	Core damage relatively low temperature of the tubes	Make water source available as carry over from the EOPs	Lowers pressure via cooling	>180
BD SAMG CHLA-3	Depressurize SG when dry	Low pressure sources of injection available	Open ADVs to lower SG pressure	Increases pressure	>180
BD SAMG CHLA-4	Restart RCP pumps (bump pumps)	Pumps have been stopped, vapor in the primary has condensed in the lower piping of the RCS	Put water into vessel by restarting pumps to provide additional water for cooling the core either before or after CD	Delays core damage by extending cooling Increases gas transport to SG tubes if CD'd	~150
Proposed SAMG to undue CHLA-3	Restore SG barrier- RCS Safety relief cooling -when tube temperature high	High temperatures in SG tubes threatening creep rupture conditions	Isolate all liquid/gas release pathways from SG; minimize the delta pressure on the SG tubes.	Lowers pressure, provides barrier	240

D.5 Defense-in-depth supported by procedures, training and plant layout

A number of plant modifications have been implemented physically in the plant but have not been modeled in the PRA study. Each one provides a form of qualitative risk margin, which is considered as a defense against an accident scenario that could challenge steam generator tube integrity. The following modifications are qualitatively seen as defense-in depth features that reduce the core damage frequency and also the thermal challenge frequency even though they have not been quantified.

- ANO2 is very well protected against station blackout events with eight hour DC battery supplies, and an additional blackout diesel generator. This diesel has a CRT start system with rapid diagnosis feedback should the diesel fail to start or run. This diesel is not under tech spec control and is therefore less likely to suffer from wear-out due to testing.

- Breakers for isolating and reloading the DC and AC safety buses are clearly marked (green and red for safety division) and with breaker numbering. Procedures for operating the breakers are available in each breaker room. Tools for opening and closing the breakers are available in cabinets near the breakers.

- Procedures for locally restarting the Turbine driven EFW pumps address over-speed trip re-latching and other protective trips on bearing temperature, vibration, etc. A bridge above the turbine permits access from the top for re-latching. Space is somewhat restricted, but should not be a significant problem for the proceduralized repairs. The EFW system is contained within a flood proof chamber, has virtually no combustible material in the room, and has multiple braces for earthquake protection.

- Ground fault detection reduces the potential for loss of the DC buses due to insulation failure within the DC safety bus systems. This has a large impact on the frequency of loss of the DC bus due to ground faults that lead to shorts on the buses. Since both buses are floating, measurements of the ground insulation quality on all systems in operation permits rapid detection, localization and correction of grounds on one bus before the appearance of a ground on the other bus. Expected ground fault correction is within a shift with a maximum of about 2 shifts. Thus, the frequency of loss of the DC bus could be reduced somewhat. At least a factor of ten for T10 & T11 sequences.

- Plant modifications reduce the likelihood of a plant trip when the DC bus is lost. (The PRA assumes that a trip occurs, because other mechanisms could cause the trip). Modifications have been introduced into the simulator.

D.6 Summary

Key points from the HRA are:

The recently installed plant modifications, and use of EOPs add additional defense-in-depth protection against SG tube ruptures that are initiated by either plant transients or core damage accidents.

These defense-in-depth features lower LERF. The degree of reduction is very sensitive to the dependencies between multiple recovery actions.

The Delta LERF is quantifiable considering the change in tube strength over the cycle alone, but remains within the acceptable regions because the initial LERF is low.

The SAMGs provide the staff with actions that can mitigate the effect of a tube failure due to creep rupture (e.g., RCS depressurization). Also permitted is isolation of the SGs to restore a barrier prior to a tube failure.

Attachment E
Review/Revision of the ANO-2 Rev-1 Cutset Results
and
Application of Additional Operator Recoveries

Introduction
A detailed description of the process used to review and apply additional operator recoveries to the ANO-2 Rev-1 cutsets is provided in this attachment. The effort involved the review and application of additional operator recoveries to the Loss of DC initiated cutsets. The recoveries were applied via the use of Human Recovery Rule (HRA) post-processing rule files. The rule files are implemented via the QRECOVER code, which is part of the EPRI Risk and Reliability Workstation, in a manner consistent with that done as part of the development of the ANO-2 Rev-1 PSA results. The steps in this process are discussed in detail below. .

Create ANO-2 Rev-1 Cutset File with Sequence Flags:

(1) Started with the ANO-2 Rev-1 Cutset File and associated Basic Event (.be) and Type Code (.tc) Files provided in Ref. 7:

@a2r1com.cut	5/10/98	10:50 pm
ANO2r1ps.be	11/4/97	10:58 am
ANO2r1ps.gt	6/5/97	10:31 am
ANO2r1ps.tc	1/1/97	01:00 am

Confirm that CDF reported in ANO2r1.cut file (i.e., (i.e., 2.084E-5) uses failure probabilities bases on the ANO2R2ps.be and .tc files by LOADing the .be file into .cut file. When this was done, the CDF remained unchanged..

(2) Started with the above cutset file and associated .be and .tc files. Using the EPRI Risk and Reliability (R&R) Cutset Utilities program, the cutsets were sorted so that initiators occur as the first term in each cutset. Again, using Cutset Utilities, the cutset file was split into modules according to class. Each cutset is marked with a class designation which is the accident sequence name (e.g., AU, TBF, etc.). Thus, the split operation created a cutset module for each sequence. The modified cutset was saved as A2R1com.cut. Note that this file remained associated with ANO2r1ps.be and .tc.

(3) Revised symbol field in basic events T10, T11, T12, T13, T14, T15 from "i" to "I" in file ANO2r1ps.be. This change flags these events as initiators, but has no impact on the probabilities of these events nor on the cutset results. Saved the revised file ANO2r1ps.be with the name A2R1com.be.

(4) Associated A2R1com.cut with A2R1com.be and saved the .cut file. Saved A2R1com.cut, .be, .tc, and .gt and zipped these files into **A2R1com.ZIP**. The zipped file is included in Attachment A of this calculation.

(5) Started with A2R1com.cut, .be, .tc, .and .gt files. Saved A2R1com.cut as A2R1com1.cut. In this new cutset, created new module called "TOTAL" and deleted all sequence modules (i.e., all other module cutsets). Saved revised cutset file A2R1com1.cut. This operation

assures that the A2R1com.cut format is preserved in A2R1com1.cut. Appended each of the sequence module cutsets from A2R1com.cut into the TOTAL module of A2R1com1.cut. This recreates the original @a2r1com.cut file, except that each cutset has a new term, the sequence flag name. Saved A2R1com1.cut. Added events AU, AX, MU, MX, RBF, RBU, RBX, RU, RX, SBF, SBU, SBX, SU, SX, TBF, TBX, TQBF, TQBU, TQBX, TQU, TQX to A2r1com.be. The value of each was set to 1.0 and each was described as a sequence flag. Saved file as A2R1com1.be. Associated A2R1com1.cut with A2R1com1.be and LOADed the BE file into the .CUT file. Cutset CDF was 2.084E-5 (as expected, the same as @a2r1com.cut).

(6) Set each of the sequence flags to TRUE. As expected, this had effect on the cutset CDF. Saved file as A2R1com1.cut. A2R1com1.cut was saved. Files A2R1com1.cut, .be, .tc, .and .gt and zipped these files into **A2R1com1.ZIP**. The zipped file is included in Attachment A of this calculation.

Initial Revisions to ANO-2 Rev-1 Cutset File

(7) The Loss of 2D11 Bus (T10) and the Loss of 2D12 Bus (T11) initiated cutsets result in the loss of RCS depressurization capability and thus in the high pressure core damage events. The T10-initiated cutsets were noted to be an important contributor to CD; whereas, the T11-initiated cutsets were noted to be much less important CD contributors. Although this trend may be valid, in order to assure conservative CDF results, the T10 initiator frequency in A2R1com1.be was doubled from 3.94E-4 to 7.88E-4/rx-yr and the T11 initiator frequency was set to FALSE. Since operator actions associated with the recovery of T10 and T11 are essentially identical, this treatment is conservative, increasing the ANO-2 CDF value from 2.084E-5 to 2.853E-5/rx-yr.

(8) Another apparent inconsistency associated with the ANO-2 Rev-1 cutsets was the presence of events PMM2302FTC, PMM2303FTC, PMM2306FTC. Each of these events represents the failure open of a TBV; each event was modeled to lead to depressurization of the SGs and closure of both MSIVs; in addition, these events were erroneously modeled to result in the failure of EFW to feed one SG. In order to correct this error and to assure conservatism in the cutset results, all of these events were set to TRUE and the cutsets were then subsumed. This change increased the ANO-2 CDF value from 2.853E-5 to 3.388E-5/rx-yr. After the subsume, the TRUEs was removed from the events and their nominal values were set to 1.56E-02 in the A2R1com2.be file. This value is intended to recognize the fact that the remaining events PMM2302FTC, PMM2303FTC, and PMM2306FTC are equivalent to failure of the feed valves to the other SG, i.e., equivalent to the occurrence of QMM2SGAP7A or QMM2SGBP7A or QMM2SGAP7B or QMM2SGBP7B (which represent failure of the feed valves to a single SG). This change revised the ANO-2 CDF from 3.388E-5 to 2.817E-5/rx-yr. The revised A2R1com2.be file was saved. This process yields a conservative overestimation of the CDF, because many cutsets are non-minimal. As a result of LOADing the revised A2R1com2.be into cutset A2R1com1.cut, the ANO-2 CDF changed from 3.388E-5 to 2.817E-5. Cutset file A2R1com2.cut saved. Files A2R1com2.cut, .be, .tc, and .gt and were zipped into **A2R1COM2.ZIP**. The zipped file is included in Attachment A of this calculation.

Preliminary Identification of Recovery Actions

(9) Cutset file A2R1com2.cut was reviewed to identify and mark operator recovery events. The review was done manually. Cutsets above 1E-9 were reviewed individually; cutsets below that value cursorily reviewed for recovery events. The following events were identified and marked as a recovery events in the cutset file.

ACXTP7BMN2	MANCSIN	OPER-5A	P7AMANRECD	SWECPREC
AFWFEEDREC	MANOSPREBD	OPER-8	P7BMANREC	SWECPRECD
AFWFEEDRES	MANOSPREC	OPER-10	P7BMANRECD	SWSWINGRES
CSXTIE	MANOSPREC2	OPER-10A	QCSTKXFER	T7REC
EDGAACREC	MANOSPRECD	OPER-15	QCSTKXFERR	
EDGAACRECR	MANOSPRECS	OPER-17	QHF2REFILL	
HOTLEGINJS	OPER-1	OPER-18M	SGISOLREC	
HPSISWING	OPER-2	OPER-18MI	SHPSISWING	
LPSPRAY	OPER-3	OPER-18RM	SUMP_REC	

After marking the recoveries, A2R1com2.cut was saved as A2R1com3.cut. The unmodified A2R1com2.be, .tc, and .gt files were renamed to A2R1com3.be, .tc, and .gt. Files A2R1com3.cut, .be, .tc, and .gt and were zipped into **A2R1com3.ZIP**. The zipped file is included in Attachment A of this calculation.

Identification of Cutsets Containing One or More "Run" or "Mission" Failure Events and/or a Battery Discharge Event

(10) The EOOS 2LOSP.CAF rule file (10/19/99 12:57) provided in Reference 15 identifies most run failures; this identification process is part of the LOSP recovery process. By deleting all but the "run" or "mission" failures" in the file and changing all gates to an OR gate, this file (renamed A2S&R.CAF) was used to generate a list of a large number of "run" failures. Quantifying this fault tree file created a list of run failures in the model. This list was copied to a QRECOVER rule file, RUNREC.txt, which follows.

```
**RECOVERY RULES**
**MAX RECOVERIES** 100
**RECOVERY** RUN 1.0
QMM2TRANAF
EDG2DG1XXF
EDG2DG2XXF
SMM2PRRCFB
SMM2PRRCFA
SMM2PRRCLB
SMM2PRRCLA
EMM2CCFDGF
SMM22P4AXA
QMM2TRANBF
LMM2VUC1AF
LMM2VUC1BF
SMM2SWLPII
SMM2SE2P4A
HMM2PASMPB
HMM2PASMPA
```

EMM2DG1FXF
EMM2DG2FXF
YMM2CSPMBF
YMM2CSPMAF
SMM22P4BXF
QMM2PMSUCF
SMM2ECCS1F
SMM2ECCS2F
QAV200798R
QAV200714R
DMM202D32F
DMM202D31F
YMM2CSBXXF
YMM2CSAXXF
DBC202D34F
QMM2CSTNKF
HMV251042K
DMM202D12F
DMM202D11F
HMV251031K
EMM2B5XXXX
EMM2B6XXXX
HMM2CCF002
STF2SCRNSP
DMM202D26F
DMM202D27F
QCD227B1FR
ECD22408XR
DCD200224R
DCD200123R
DCD22404XR
DCD22304XR
ECD22308XR
QCD227B2FR
QSV200798R
QSV200714R
HCV2SI10BK
HCV2SI-7BK
HCV2SI10AK
HCV2SI-7AK
HCV22SI12K
EMM2CCFFXF
YMM2CCFPMF
EMM2B53XXX
EMM2B51XXX
EMM2B62XXX
EMM2B63XXX
EMM2B64XXX
EMM2B61XXX
EMM2B52XXX
EMM2B54XXX

ECB262B5XR
ECB254H3XR
ECB263K5XR
ECB264D5XR
ECB253K3XR
ECB252B5XR
SMM2CCFMDP
ERE2A4LXXK
ERE2A3LXXK
DMM2CCFBTF
EMM2A3XXXX
EMM2A4XXXX

This rule file was applied to A2R1com3.cut to add the event "RUN" to each cutset containing at least one run failure. Event RUN was set to TRUE. In order to assure that event RUN was added to every cutset containing a mission failure, MAX RECOVERIES was set to 100 in the recovery rule file. After executing QRECOVERY, the CDF remained 2.817E-5/rx-yr, as expected. A2R1com3.cut was saved as A2R1com4.cut.

(11) A second QRECOVER rule file, BTDREC.txt, was created to add flag event "BTD" to all cutsets containing either battery discharge event DBT2DSCD11 or DBT2DSCD12. In order to assure that event BTD was added to every cutset containing a battery discharge event, MAX RECOVERIES was set to 100 in the recovery rule file. The QRECOVER BTD rule file, BTDREC.txt, follows:

RECOVERY RULES
MAX RECOVERIES 100
RECOVERY BTD 1.0
DBT2DSCD11
DBT2DSCD12

After executing QRECOVER on A2R1com4.cut, event BTD was set to TRUE. After executing QRECOVERY on A2R1com4.cut, the CDF remained 2.817E-5, as expected. Files A2R1com3.be, .tc, and .gt were saved as A2R1com4.be, .tc, and .gt. Events RUN and BTD were added to A2R1com4.be with values of 1.0 and the file was saved. Files A2R1com4.cut, .be, .tc, and .gt were zipped into **A2R1com4.ZIP**. The zipped file is included in Attachment A of this calculation.

Combination of Operator Actions Related to AC Power Restoration

(12) Operator recovery events MANOSPREBD, MANOSPRECD, MANOSPREC, MANOSPRECS, and MANOSPREC2 model operator actions to manually open/close SUT breakers; Operator recovery event EDGAACREC models operator actions to use the Alternate AC Diesel Generator (AACDG) to restore on-site power. These events were combined to reflect the fact that the recovery of offsite power and use of the AAC DG are independent means of restoring power to the safety buses. The combination was accomplished by reducing the values of MANOSPREBD, MANOSPREBD, MANOSPREC, MANOSPRECS, and MANOSPREC2 by the value of EDGAACREC. In effect, MANOSPREBD, MANOSPREBD, MANOSPREC, MANOSPRECS, and MANOSPREC2 were redefined to be the failure to manually open/close SUT breakers and

to start AACDG. This redefinition in the .be file is possible, since event EDGAACREC or EDGAACRECR never appear in the same cutset as a MANOSPRxx event.

Old Event	Probability	New Event	Probability
EDGAACREC	0.117	Same	Same
MANOSPREC	0.13	MANOSPREC (=MANOSPREC*EDGAACREC)	0.0152
MANOSPREBD	2.80E-03	MANOSPREBD (=MANOSPREBD*EDGAACREC)	3.28E-04
MANOSPRECD	5.70E-02	MANOSPRECD (=MANOSPRECD*EDGAACREC)	6.67E-03
MANOSPRECS	3.50E-01	MANOSPRECS (=MANOSPRECS*EDGAACREC)	4.92E-02
MANOSPREC2	1.80E-01	MANOSPREC2 (=MANOSPREC2*EDGAACREC)	2.11E-02

These events were revised in A2R1com4.be and the revised .be file was saved as A2R1com5.be. The CDF for the cutset file changed from 2.817E-5 to 2.479E-05. This file was then LOADed into A2R1com4.cut and the resulting file was saved as A2R1com5.cut. The files A2R1com5.cut, .be, .gt, and .tc were zipped into **A2R1com5.zip**.

Application of Composite Operator Recovery Actions to T10-initiated Cutsets

(13) Prior to applying the composite recovery operator actions developed in Attachment D for the loss of DC initiator, operator recoveries ACXTP7BMN2, OPER-13, and OPER-15 were deleted from all T10 cutsets. After their removal, the CDF for the cutset changed from 2.479E-05 to 1.264E-04. This file was saved as A2R1com6.cut. A2R1com5.be, .tc, .gt were copied to A2R1com6.be, .tc, .gt (no changes were made; the name change was to ensure consistency with the cutset name). The files A2R1com6.cut, .be, .gt, and .tc were zipped into **A2R1com6.zip**.

(14) Except for recovery OPER-1, all remaining operator recoveries present in the T10-initiated cutsets were manually deleted from cutset file A2R1com6.cut. OPER-1 ("Operator fails to trip RCPs after CCW loss") was retained in these cutsets, since operator actions associated with this event is highly independent of other actions. The deletion of these recoveries from the T10-initiated cutsets was done in preparation for applying the composite recovery operator actions to these cutsets. The revised cutset file was saved with the name A2R1com7.cut.

(15) After removing the subject operator recoveries from the T10-initiated cutsets, the composite recoveries were applied to these cutsets. In Attachment D, two composite operator recoveries were developed for T10-initiated cutsets:
   (a) operator recovery OA-T10-STR is defined to the composite operator recovery associated early core damage scenarios; the probability of this event was developed in Figure 1 of Attachment D to be 3.31E-03.
   (b) operator recovery OA-T10-RUN is defined to the composite operator recovery associated late core damage scenarios; the probability of this event was developed in Figure 2 of Attachment D to be 5.19E-04.

These events were added to the basic event file A2R1com6.be and the file renamed to A2R1com7.be. The OA-T10-RUN was applied to all T10 cutsets containing either a BTD or a RUN flag event; OA-T10-STR was applied to the remaining revised T10 cutsets via a QRECOVER rule file developed for this purpose. A listing of this rule file is as follows:

```
**RECOVERY RULES**
**MAX RECOVERIES** 100
;
T10 RUN
**RECOVERY**    OA-T10-RUN
T10     RUN
T10     BTD
**RECOVERY**    OA-T10-STR
T10     -RUN        -BTD
```

After removing the subject recoveries and applying the composite T10 recoveries, the cutset CDF changed from 1.264E-04 to 1.235E-05/rx-yr. This file was saved as A2R1com7.cut. The files A2R1com7.cut, .be, .gt, and .tc and the rule file T10RECs.txt were zipped into **A2R1com7.zip**.

Final Identification of Recovery Actions and Application of Generic Operator Recovery OA2

(16) Attachment D indicated that it is appropriate to apply a second recovery to a cutset when only one recovery is applied to the cutset. This generic "second recovery" event was called OA2. Per Attachment D, this recovery action can be assigned a value of 0.15 when AC power is available and a value of 0.5 when AC power is not available. The latter is interpreted as the Station Blackout (SBO) condition. Since the SBO contribution to the ANO-2 CDF is only about 5% of the total CDF, for simplicity, the value of OA2 was chosen to be 0.15. This assumption is slightly non-conservative, but a sensitivity analysis ot the error indicates that it represents only about a slight (~1%) underprediction in CDF had 0.5 been used for the SBO cases.

(17) Table D-2 of Attachment D of Reference 10 provides a list of operator recovery events. Combining this list with the recoveries documented in Reference 14, the following post-initiator operator recoveries were marked as recovery events in the A2R1com7.cut file:

ACREALIGND	FHF20FWCSD	MANOSPREBD	OPER-13	OPER-3	QRECAFW
ACXTP7BMN2	HOTLEGINJ	MANOSPREC	OPER-14	OPER-5A	RHPSISWING
AFWFEEDREC	HOTLEGINJM	MANOSPREC2	OPER-15	OPER-7	SGISOLREC
AFWFEEDRES	HOTLEGINJS	MANOSPRECD	OPER-17	OPER-8	SHPSISWING
CSXTIE	HPSIREC	MANOSPRECR	OPER-18E	OPER-9	SUMP_REC
EDGAACREC	HPSISWING	MANOSPRECS	OPER-18EI	P7AMANREC	SWECPREC
EDGAACRECD	HRECIRCREC	MANSCD	OPER-18M	P7AMANRECD	SWECPRECD
EDGAACRECR	IAXTIE	OPER-1	OPER-18MI	P7BMANREC	SWREC
EDGOPER1	LHF2HE005D	OPER-10	OPER-18RE	P7BMANRECD	SWSWINGREC
EDGOPER2	LHF2HE006D	OPER-10A	OPER-18RM	QCSTKXFER	SWSWINGRES
ERECA3/A4	LPSPRAY	OPER-11	OPER-2	QCSTKXFERR	T7REC
ERECB5/B6	MANCSIN	OPER-12	OPER-3	QHF2REFILL	XREC2E28BC

After marking these events as recovery events, cutset file A2R1com7.cut was saved as A2R1com8.cut. As expected, these changes had no impact on the cutset CDF of 1.235E-05. Files A2R1com7.be, .gt, and .tc were copied to A2R1com8.be, .gt, and .tc for consistency, and the files A2R1com8.cut, .be, .gt, and .tc were zipped into **A2R1com8.zip**.

(18) Application of the generic second recovery OA2 was accomplished by modifying the QRECOVER HRA rule file associated with the ANO-2 EOOS model documented in Attachment D of Reference 10. This rule file was modified as follows in order to allow it to apply recovery OA2 to cutsets containing less than two recoveries and for which a second recovery action is applicable. The changes are as follows:

(a) MAX RECOVERIES changed from 1 to 2. This will allow OA2 to be applied to cutsets which already contain one recovery, but not to cutsets containing two or more recoveries.

(b) Deleted INJECT ALL CUTSETS**TRUE from file. Has no effect on the recovery process; thus, this input is not needed.

(c) RECOVERY TREE revised to point to ANO-2 Rev-1 model, ANO2R1.caf.

(d) Replaced all RECOVERY events with OA2 and put the replaced recovery as a NOT event in every line of that recovery event's rules.

(e) NOT MANOSPREBD was added to rules associated with recoveries MANOSPRECD, MANOSPREC2, and MANOSPRECS that immediately follow MANOSPREBD. Likewise, NOT MANOSPRECD was added to rules associated with recoveries MANOSPREC2 and MANOSPRECS that immediately follow MANOSPRECD. And, NOT MANOSPREC2 was added to rules associated with recovery MANOSPRECS that immediately follow MANOSPREC2. This NOT logic is not absolutely required to prevent OA2 from being applied correctly to cutsets containing any of these events; however, the NOT events were added to additional assurance that OA2 would be applied correctly.

(f) "%" deleted from initiators %T11, %T12, %T2 to assure consistency with ANO-2 Rev-1 model names for these initiators.

(g) Added the following new lines in the CSXTIE rule files for sequences AU, MX, SX, SBX, RBX, TBX, TQX, and TQBX:

(sequence)	YMM2CSAXXA	YMM2SWPMPB	-CSXTIE
(sequence)	YMM2SWPMPA	YMM2CSBXXA	-CSXTIE

where, (sequence) is the accident sequence name, AU, MX, SX, SBX, RBX, TBX, TQX, or TQBX. These events were inadvertently excluded from the EOOS HRA rule files, but are logically appropriate for these files.

(h) Similar to what was done for the MANOSPREC recoveries, NOT SWECPRECD was added to rules associated with recovery SWECPREC that immediately follow SWECPRECD. As with the MANOSPREC rules, this NOT logic is not absolutely required to prevent OA2 from being applied correctly to cutsets containing any of these events; however, the NOT events were added to additional assurance that OA2 would be applied correctly.

(i) Similar to what was done for the MANOSPREC recoveries, NOT P7BMANRECD was added to rules associated with recovery P7BMANREC that immediately follow

P7BMANRECD. As with the MANOSPREC rules, this NOT logic is not absolutely required to prevent OA2 from being applied correctly to cutsets containing any of these events; however, the NOT events were added to additional assurance that OA2 would be applied correctly.

(j) In order to avoid recovery OA2 from being added to T10 cutsets containing either the composite recoveries OA-T10-STR or OA-T10-RUN, a NOT T10 was added to all OA2 rules.

(k) Although not activated (i.e., a ";" was put as the leading character of each line), the following rules for AFWFEEDREC were added to the rule file:

```
;**RECOVERY**     OA2    0.15     @B01-NI
;TBF    -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDREC
;TBX    -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDREC
;TQBF   -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDREC
;TQBU   -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDREC
;TQBX   -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDREC
;
;**RECOVERY**         OA2          0.15          @B01-NI
;SBF    -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDRES
;SBU    -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDRES
;SBX    -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDRES
;RBF    -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDRES
;RBU    -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDRES
;RBX    -EDGAACREC    -EDGAACRECD  -EDGAACRECR   -AFWFEEDRES
```

The revised rule file was named HRArules.txt. This file is provided in Attachment A.

(19) This QRECOVER rule file was applied to A2R1com8.cut and the resulting cutset file was renamed A2R1com9.cut. The cutset frequency changed from 1.235E-05 to 1.118E-05/rx-yr. In addition, event OA2 with a value of 0.15 was added to the A2R1com8.be file and this file was renamed A2R1com9.be. Files A2R1com8.gt and A2R1com8.tc were copied to A2R1com9.gt and .tc, for completeness. Then, files A2R1com9.cut, .be, .gt. and .tc together with the rule file HRArules.txt were zipped into **A2R1com9.zip**.

Attachment F
Arkansas Nuclear Unit One, Unit 2
External Events Investigation
For Steam Generator Tube Integrity Risk Assessment

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The risks from external events at Arkansas Nuclear Unit One, Unit 2 (ANO-2) have been investigated using the IPEEE process (Reference F1). This process has as its objectives:

- Develop an appreciation of severe accident behavior,
- Understand the most likely severe accident sequences that could occur at the plant under full power operating conditions,
- Gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and
- If, necessary, reduce the overall likelihood of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

In addition to earthquakes, internal fires, high winds and tornadoes, external floods as well as transportation and nearby facilities accidents were evaluated as part of the IPEEE process. The conclusion from the IPEEE process is that ANO-2 has a relatively low risk from external events.

The EPRI Seismic Margins Method as defined in NUREG-1407 (Reference F2) and modified by Generic Letter 88-20 (Reference F3) was used for the seismic evaluation. ANO-2 is located in Arkansas and its principal structures are founded on rock. This evaluation verified that the equipment, tanks, distribution systems, structures and relays are able to withstand a 0.3g Review Level Earthquake at the plant and still provide for its safe shutdown.

Several issues relative to plant safety from the seismic analysis were identified; however, none of these represent any adverse operability issues at ANO-2. In addition, Reference F1 indicates that all open issues resulting from the IPEEE process are or will be resolved.

It is not possible, based upon the IPEEE evaluation described above, to make quantitative estimate regarding steam generator tube integrity (SGTI) risks from external events. The qualitative conclusion of relatively low risk from external events from the IPEEE process implies that the contribution of external events to the ANO-2 core damage frequency is relatively low.

Since severe accident induced STGI risk is directly related to the core damage frequency, the contribution of external events to SGTI risk should also be relatively low. This might not be true if the fraction of core damage events that lead to SGTI risk for external events is much higher than for internal events. However, based on the features and capabilities at ANO-2 as seen from the assessment of their availability for internal events, this is not the case.

For example, the emergency feedwater system at ANO-2 can function without bulk ac power through operation of the turbine-driven pump and it is well designed and maintained to maximize its availability to inject feedwater during external events. In addition, there is a separate stand-alone capability using an auxiliary feedwater system with its own power supply, the Alternate Ac Diesel Generator (AACDG), to ensure continued feedwater supply for the steam generators.

Based on the combination of these two systems, feedwater supply availability will not be more significantly affected by external events than internal events. Thus the fraction of core damage events that lead to SGTI risk should not be much higher for external events than for internal events. Therefore, the conclusions of the IPEEE process that the contribution of external events to the ANO-2 core damage frequency is relatively low should also be valid for the frequency of severe accident induced thermal challenges to the steam generators. Thus in estimating the thermal challenge frequency for ANO-2 from all events, the qualitative IPEEE results for external events are accounted for as an add-on 50% contribution to the estimate of internal events. This is consistent with the understanding that the contribution from external events is relatively low. The difference between this ballpark figure for external events and an estimated contribution that would be based on quantified PRA models is treated in the overall uncertainties associated with the thermal challenge frequency.

References

F1. Lach, D.L., Robinson, T. and Moser, M., Summary report Of Individual Plant Examination Of External Events (IPEEE) For Severe Accident Vulnerabilities For Arkansas Nuclear One, Unit 2, May 1996.

F2. U.S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," USNRC Report NUREG-1407, June 1991.

F3. Generic Letter 88-20, Supplement 5, "Individual Plant Examinations of External Events (IPEEE) for Severe Accident Vulnerabilities," U.S. Nuclear Regulatory Commission, August 1995.

Attachment G
Arkansas Nuclear Unit One,
Unit 2 Human Reliability Assessments
Associated with Maintenance of SG Pressurization
and RCP Loop Seals

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G.1 Assessment of Recovery Actions for Use in Evaluating SG Tube Integrity at ANO-2

G.1.1 Defining the Accident Context

All internal risk sequences in the ANO-2 PRA Model Rev-1 have been reviewed to define those with the potential for entering a core damage state with a HIGH RCS pressure, and a DRY steam generator. The likelihood of the HIGH-DRY condition depends on how the operators use the emergency operating procedures (EOPs) to compensate for key hardware failures. The likelihood of a low SG pressure state at the time when hot gases can naturally circulate, if the RCS boils dry, depends on the way operators use both the EOPs and severe accident management guidelines (SAMGs). The outputs of this assessment support estimates of the fraction of time that the risk sequences result in a LOW steam generator pressure at the time of natural circulation of gases in the primary. They also support estimates for the fraction of time that the RCP seal loops will be cleared given a high/dry/low SG pressure condition.

Thus, the review of existing cutsets in Rev 1 of the PRA conservatively selected those sequences with the potential for a HIGH RCS pressure, and DRY SGs at the time of core damage. This eliminates sequence candidates for thermal challenge to the SG tubes that involve medium and large LOCAs, and transients with success in the feedwater system. The primary transient sequences remaining consist of those involving loss of feedwater that stem from loss of bus trips, station blackouts, small LOCAs and hardware failures.

An Expanded Event Tree (Figure 6 in the main report) was constructed to define the key conditions for assessing additional TI-SGTR Issues. Event Tree addresses success and failure of systems and actions by the operator that help define the likelihood of possible plant states given core damage and hot gases circulation in the RCS. This condition can heat up the SG tubes under enough pressure to cause creep rupture. The key operator actions are those that impact SG Pressure, RCS Depressurization, and RCP Loop Seal Conditions. Combinations of operator actions define these conditions and their quantification supports the likelihood of a thermal challenge to the SG tubes.

The existing core damage cutsets do not account for operator actions taken to mitigate the consequences of core damage. These operator actions are defined by the ANO-2 Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs). The EOPs direct actions prior to core damage; and, the SAMGs provide guidance in managing an accident involving core damage. The loss of feedwater and functional recovery EOPs are "precursors" to the SAMGs and in theory pre-condition the operators to the importance of maintaining a high SG pressure in a core damage event. Then the SAMGs should help to further reduce the probability of a SG depressurization (other than those due to hardware failure).

The basic logic of the expanded tree applies to all sequences involving loss of the secondary heat source for an extended period. Some proceduralized recovery paths may be limited by the context of the accident sequence, that is in the case of station black out actions requiring power would be set with HEPs equal to 1.0. However, the quantification is based on selection of the upper and lower bound HEPs for the human action(s) and estimate of the mean value of the log normal distribution for those end points.

G.1.2 Qualitative HRA Issues

Application of the EOPs [References G1 and G4] in response to an initiator provides the operators with guidance and priorities for establishing cooling success paths, which will prevent core damage by maintaining sufficient cooling for the decay heat levels in the core. The operators routinely use the ADV's to remove heat until of the feedwater success paths can be established. ANO-2 has five success paths for heat removal described in the EOPs. These success paths are:

1. Motor Driven EFW
2. Turbine Driven EFW
3. Auxiliary Feedwater
4. Condensate System
5. Once Through Cooling

Success paths1, and 3, use electrically driven pumps (EFW and AFW) that depend on AC and DC power supplies, and can be operated at manual locations within the plant. Success path 2 uses the turbine driven pump that is powered by steam from either steam generator. Its controls are supported by DC power if available, but can be operated without DC power locally. Success path 4 requires DC, AC and operator actions to reset protective trips and to line up the MFW/Condensate pathway. Success path 5, RCS depressurization, requires AC power for HPSI injection, and at least one division of DC power to open the high point vent pathway into the containment. The electrical system includes three large diesels dedicated to Unit 2 (two safety and one auxiliary), diverse offsite power sources that power the main and startup transformers. Connection of a source of power to the supply bus from any source of AC power can be locally accomplished locally by manually aligning breakers. The batteries provide more than eight hours supply to the DC safety buses.

Of these success paths, all but success path 4 are considered in the initial fault tree models and are assumed to be failed at the time of core damage in the sequences selected as thermal challenge candidates. In the case of success path 4, no credit is taken.

If the actions are applied as indicated in the EOPs, success of any hardware pathway will avoid core damage. Thus, the operators using the ADV's or Steam Dumps to maintain cooling have significant time to recover feedwater in scenarios dominated by run failures, and use once through cooling options if there is not enough time to recover the feedwater systems. The use of the ADV's has another significant benefit of avoiding a large number of MSSV challenges. Thus, operators use of the ADV's as stated in the EOPs is considered in the event tree of Figure 6.

The reasons for using ADVs /steam dumps include:

- Controls heat removal from about 1000 psi to lower (below the set point of the MSSVs)
- Avoids MSSV lifts

Reasons for not using ADVs/steam dumps include:
- Their continued use can cause the cooldown transient to exceed vessel cool down rate limit
- Uses SG inventory slightly faster than MSSV relief valves
- Depressurizes the secondary side to promote alternative sources of low-pressure feedwater injection

Thus, the operators are likely to initially use the ADVs. This results in limiting the number of MSSV lifts.

If one of the hardware paths becomes successful after core damage, but before tube damage this switches a high primary pressure/ low secondary pressure to a medium primary / medium secondary pressure when the feedwater is restored, eliminating the thermal challenge condition. This type of recovery is not considered nor credited in this evaluation even though the severe accident management guidelines (SAMGs) support continued efforts to restore secondary cooling. Thus, the key unique features of the SAMGs that change the pressure differential across the tubes are to:

- Depressurize the Steam Generators to enable feedwater injection in Candidate High Level Action (CHLA); this increases creep rupture potential if applied in the wrong way), and
- Restart RCS pumps to sweep non-condensable gases and clear loop seals to promote natural circulation CHLA (this increases creep rupture potential if applied in the wrong way).

The key steps in evaluating the impact of operator actions on the potential for high/dry/low conditions was accomplished by first reviewing the ANO-2 PRA study for sequences that threaten SG tube integrity under severe accidents. Then, for representative sequences, manual actions were identified that can be performed locally to maintain cooling and avoid SGTI challenges.

To verify the ability to perform actions with EOP talk through and simulator observation of the T10 initiator verified steps used to manage high-dry sequence. Also the use of SAMGs was examined using a talk through method for key risk sequences. The approaches for integrating the HRA results into calculations of LERF and delta LERF using the current study structure were modeled in the inputs to Figure 6.

G.2 Quantification of Key Actions

The qualitative Performance Shaping Factor (PSF) input values can be transformed into quantitative P1, P2, & P3 values through calibrated models. PSFs can be applied at the individual error level or at the overall error probability as shown in equation G1 for the probability of operator success, where xxOAxx represents a generic operator functional action.

Pr (xxOAxx) = 1-HEP (context, power availability, MMI, EOP clarity, training, SRO evaluation, etc.). Eq. G1

The formulation can be calibrated using published results for mixed PSF qualitative combinations to Human Error Probability (HEP) cases that are provided in reports such as ASEP NUREG-4772 [Reference G2] for errors of omission (e.g., skipping procedure steps), and EPRI TR-100259 [Reference G3] for cognitive errors. The basic HEP used here is from ASEP (NUREG-CR/4772), HEP = .03, when there is training, a cue, and procedures are understood. The following adjustments have been made considering the difficulty of the updated procedures, observations of a sequence in training simulator and considering the application to a wide range of scenarios. Thus, end points of the distributions are selected, and the associated mean value is determined by simulation.

Three human actions were considered as the important ones for quantification.

G.2.1 Operator fails to use ADVs (or steam dump) to control SG pressure and cooldown

Use of ADVs (or steam dump valves) is part of the EOP application. Considering the range of accident scenario contexts for this action, the use of procedures, possible conflicts and the observation of simulator responses to typical transients, the following end point parameters have been selected to bound HEP for the range of sequences contained in the list of potential thermal challenges.

Based on ANO-2 Emergency Operating Procedure (EOP) 2202.009 HR-1 SG Heat Sink without SIAS Functional Recovery [Reference G4], a screening value was assigned by selecting the following Lognormal distribution parameters:

Parameter	Value
10 percentile	0.0100
90 percentile	0.1000
Mean value	0.0466

The following distribution describes the HEP for operators failing to use ADVs to control SG pressure and cooldown rate.

Percentile	HEP
0.0%	0.001
2.5%	0.005
5.0%	0.007
50.0%	0.031
95.0%	0.136
97.5%	0.183
100.0%	0.953

G.2.2 Operator clears a RCP loop seal, with no heat sink available

This action is found in the SAMGs. Loop seal Clearing with the RCP is recommended only when the heat sink is available, which is a key for enhancing heat removal. The error is in applying this action when there is no heat sink. In this case it promotes more rapid heat up of the SG tubes.

Based on ANO-2 Severe Accident Management Guidelines [Reference G5], the CHLA associated with restarting the RCPs has the following note at the beginning of the procedure.

NOTE
A viable heat sink (SG with makeup available/secondary inventory) must be available for this CHLA to be considered.

The key error is in skipping the note and performing the action. The standard error probability from ASEP (0.03) is used for this action, and the following parameters are defined to consider the range of conditions associated with the different sequences considered.

Parameter	Value
Geometric Mean	0.03
Geometric Std. Deviation	2.0
Mean value	0.0407

The distribution is described as follows:

Percentile	HEP
0.0%	0.0024
2.5%	0.0081
5.0%	0.0101
50.0%	0.0316
95.0%	0.1010
97.5%	0.1289
100.0%	0.4855

G.2.3 Operator fails to attempt to isolate SGs

This action is found in the SAMGs [Reference G5] and is preconditioned by the EOPs. SAMG BD/CC CHLA 3 (and other CHLAs for other plant conditions) permits operators to depressurize the SGs to promote heat removal. The error is in applying the procedure before the low-pressure feedwater system is available, and when natural circulation of hot gas in the primary is under way. In this case SG pressure can't be restored when before the tubes heat up due to hot gas natural circulation. SG depressurization promotes a large pressure differential across the tubes and creates the conditions for possible of creep rupture. Thus, this is terminated when any of the following criteria apply:

Termination of the CHLA action is considered if depressurization could lead to creep rupture of the SG tubes. Conditions which could lead to creep rupture of the steam generator tubes are ALL of the following

- RCS temperature above the core is >900F
- SG to be or being depressurized has a level lower than 70" (wide range) AND no secondary makeup available
- RCS pressure is above about 1000 psig

To provide a generic estimate the following parameters are applied. This is based on the fact that this procedure is controlled from the technical support center and not the control room. The cautions include clear Cues for terminating the action. One issue is that the functions of isolating must be seen as equivalent to not depressurizing.

Parameter	Value
10 percentile	0.0300
90 percentile	0.3000
Mean value	0.1393

The following distribution describes the distribution for operators failing to isolate the SGs when temperature in RCS increases rapidly.

Percentile	HEP
0.0%	0.003
2.5%	0.016
5.0%	0.022
50.0%	0.096
95.0%	0.407
97.5%	0.516
100.0%	0.999

G.3 Summary

Key points from the HRA are:

The recently installed plant modifications, and use of EOPs add additional defense-in-depth protection against SG tube ruptures that are initiated by either plant transients or core damage accidents.

These defense-in-depth features lower LERF. The degree of reduction is very sensitive to the dependencies between multiple recovery actions, and use of SAMGs

The SAMGs provide the staff with actions that can mitigate the effect of a tube failure due to creep rupture. (e.g., RCS depressurization). Also permitted is isolation of the SGs to restore a barrier prior to a tube failure.

G.4 References

G1 ANO-2 Procedure 2202.006, Revision 004-01-0, "Loss of Feedwater"

G2 NUREG/CR-4772, Swain, A. D., "Accident Sequence Evaluation Program: Human Reliability Analysis Procedure"," US NRC Washington D.C., February 1987

G3 EPRI TR-100259, "An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment," Electric Power Research Institute, Palo Alto, CA, June 1992

G4 ANO-2 Procedure 2202.009, Revision 004-02-0, "HR-1 SG Heat Sink without SIAS"

G5 ANO-2 Severe Accident Management Guidelines, Revision 3/1/2000

Attachment H
MSSV Failure to Reclose Probability Estimates for SBO and Loss of DC Initiated Severe Accident Sequences at ANO-2

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A severe accident sequence involving high primary pressure and failure of all feed water systems is predicted to produce multiple Main Steam Safety Valve (MSSV) lifts before and after postulated core damage. This sequence is important because, if the MSSV fails in an open position, the steam generator pressure will be reduced to ambient conditions. The resulting increase in differential pressure across the steam generator tubes increases the likelihood that one or more of them will fail.

MSSV reliability database

To support estimates of MSSV failure to re-close, the database on licensee event reports maintained by ORNL was sorted on plant trips to identify the potential for challenges that lift the MSSVs and to look for cases where the valves failed to re-close. The selection of transient events maintained by ORNL included all PWR transients that caused plant trips in US plants over ten years from 1986 to 1996. The selected events were examined for multiple MSSV lifts during a single transient. The documented openings represent one opening of one or more relief valves in each transient; 1286 transient events were identified, and the database is reported in Appendix C of Reference [H1].

In reviewing the 1286 transients, 218 lifts and re-closures with one failure were identified in the LERs and verified by discussion with plant personnel. The 218 lifts and re-closures were produced in 94 transients at 24 plants. The distribution of multiple valve lifts ranged from four cases of 13 per transient to twenty -one cases of 2 lifts and re-closures per transient. In addition 316 other transients with a high potential for causing a lift of the MSSVs were identified in the sample of 72 plants (94 of these were in the sample of 24 plants where MSSV lifts were documented). These data were used to estimate a median value, the lower limit, and an upper bound, as described below.

MSSV failure event description

One event as been identified as a failure of the MSSV*. On May 19, 1996, a malfunction in the feed water control circuitry caused a prompt reduction in speed and corresponding output of the Main Feed Pump A at ANO-1, resulting in a reactor trip on high reactor pressure. Six of the eight main steam safety valves on Steam Header B opened as designed, but one of the six failed to close after steam pressure dropped. The pressure in Once-Through Steam Generator A did not reach the main steam safety valves lift set point because of the reduced inventory in the steam generator as a result of Main Feed Pump A speed reduction. This resulted in a lower-than-normal steam generator initial pressure after the reactor trip. This also allowed more time for the turbine bypass valves to open and reduce the peak pressure. In accordance with plant procedures, the operators isolated Steam Generator B, allowing it to boil dry through the open safety valve. Following reactor trip, normal feed water flow was lost. Emergency feed water actuated as

* Failure documented in NRC Information Notice 96-61: "Failure of a Main Steam Safety Valve to Reseat Caused by an Improperly Installed Release Nut".

designed and provided a decay heat removal path through Steam Generator A and the condenser. Later the condenser became unavailable, and decay heat removal proceeded through the atmospheric dump valves.

After shutdown the stuck-open safety valve was identified, it was re-seated and gagged shut. Steam Generator B was then filled using emergency feed water approximately 5 1/2 hours after it was allowed to boil dry. From this point, recovery of the plant proceeded without complications. Analysis of the effects of the temperature transient on the reactor vessel and on the Steam Generator B shell and tubes indicated that this equipment had not been adversely affected and the plant could be returned to operation. The steam safety valve failure is attributed to improper maintenance. A castellated nut at the top of the valve stem is held in place by a cotter pin through the nut and stem. The cotter pin had not been properly engaged with the nut and, as the valve discharged, the valve vibration permitted the nut to rotate down the threaded stem and come to rest on the top of the valve-lifting lever. When the valve was closing, the nut against the lever prevented the stem from dropping, holding the valve open.

Failure rate estimates for MSSVs from operational history

Based on the event data evaluation the following statistically based estimate is provided for the MSSV failure rate per lift. The central estimate is determined from the formulation described in Reference [H2] for one observed failure in the sample population.

$$\lambda = 4.5E\text{-}3/\text{demand}$$

A lower bound on this estimate of the expected failure rate is estimated by considering the fraction of transients for the remaining plants where the lifts are expected to have occurred as counted for the plants with good documentation, but because the lifts were not documented are unknown.

$$\lambda_{LB} = 2.5E\text{-}3/\text{demand}$$

An upper bound can be determined by assuming that the distribution for this failure rate is log normal. Based on the properties of the log normal distribution, the ratio of the lower to the median equals the ratio of the median to the upper. The estimate is,

$$\lambda_{UB}=8.4E\text{-}3/\text{demand}$$

This estimate clearly applies to the first lift of one or more valves. If the number of lifts on a valve exceed one, then the effects of multiple cycle wear must be considered. The first lift includes maintenance errors and standby failures that effect valve reliability. Once the first lift is successful, the subsequent lift failure rate is much lower for subsequent valve actuations, as demonstrated during valve testing by the manufacturers and EPRI. The failure rate estimated above provides a basis for estimating the probability of valve failures on the first lift of any MSSV that opens.

A failure to re-close per lift rate averaged from many IPEs is 7.E-3/ lift. Thus, this more detailed evaluation demonstrates about a factor of two reduction in central estimate of the per demand failure rate for the MSSVs, and a tighter distribution on the uncertainty.

MSSV Failure Probability Due to Repeated Operation

The operational transient data suggest that there is a probability of failing to re-seat after the first lift of 4.5E-03 per demand, due to maintenance errors. For a two-loop plant the cumulative probability of failure to re-seat after the first lifts is 9.0E-03/RY. In addition, the MSSVs could be challenged many times in the course of a severe accident with high pressure and loss of all feed water, as the steam generators are drying out (such as an event with a T10 initiator or a station blackout event). Even though the valves would be functioning within design parameters, there is a possibility that they would fail to re-seat due to excessive wear. The valve manufacturers (Crosby and Dresser) have obtained considerable test data on MSSVs. An analysis of data taken by EPRI for the Westinghouse Owners Group [H3] reveals that 234 valve cycles were observed without any failures. One valve (a Crosby 6R10 MSSV) was tested over 90 times without failure, and showed relatively little evidence of wear. Between Crosby and Dresser, more than 1400 tests on MSSVs under prototypical conditions (including the 90 tests noted above) have been carried out without failure. One of Dresser's valves was tested more than 200 times without failure [H4]. Since there were no failures, an estimate of the failure rate for any given cycle can be obtained from the following formula from Reference [H2]:

$$P_{fail} = 0.55/1400 = 3.93\text{E-}04 \text{ per cycle.}$$

MSSV failure probability for station blackout sequences

The sequences analyzed with MAAP 4.0.3, discussed in Appendix B of Reference [H1], resulted in about 85 MSSV cycles per loop for a station blackout sequence. Based on this assessment, the cumulative failure probability from 85 lifts of the MSSV with the lowest set point in its loop is estimated to be, for a station blackout sequence,

$$P_{fail} = 4.5\text{E-}03 + 1 - P_{success} = 4.5\text{E-}03 + 1 - (1 - 0.000393)^{85} = 0.0374.$$

The likelihood that there would be no failures between the two loops is estimated by

$$P_{success} = (1- 0.0374)^2 = 0.9266.$$

The failure probability is thus 0.0734.

MSSV failure probability for T10 sequences

For T10 sequences, power is available, so that relatively few challenges to the MSSVs would be made before the operators would control the steam generator pressure, either using the turbine bypass valves or the atmospheric dump valves. Assuming (conservatively) that five cycles would occur for each valve with the lowest set point, the failure probability for each MSSV is given by

$$P_{fail} = 4.5E\text{-}03 + 1 - (1 - 0.000393)^5 = 0.00646.$$

For both loops combined, the likelihood that there would be no MSSV failures to re-seat is thus

$$P_{success} = (1 - 0.00646)^2 = 0.9871.$$

The failure probability is then 0.0129.

References

H1. E.L. Fuller, E.T. Rumble, G.W. Hannaman, and M.A. Kenton, "Assessment of Risks from Thermal Challenge to Steam Generator Tubes During Hypothetical Severe Accidents: Diablo Canyon as an Example Plant," EPRI TR-107623, Vol. 2, May 1998.

H2. M. M. R. Williams and M. C. Thorne, "The Estimation of Failure Rates for Low Probability Events," Progress in Nuclear Energy Vol. 31, No. 4, pp. 373-476, 1997.

H3. Continuum Dynamics, Inc. "Summary of Crosby Main Steam Safety Valve Test Program Results," Electric Power Research Institute Project 2233-21, Palo Alto, CA 1989.

H4. D. K. Sharma to J. W. Richardson, "Dresser Memo on SV Reliability," Dresser Industries Internal Company Memorandum, April 15, 1996.