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Ref: 10CFR50.90

CPSES-200200229
Log # TXX-02023
File # 00236

February 4, 2002

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) DOCKET NOS. 50-445 AND 50-446 SUPPLEMENT ONE TO LICENSE AMENDMENT REQUEST (LAR) 01-14 REVISION TO TECHNICAL SPECIFICATION (TS) 5.5.16 CONTAINMENT LEAKAGE RATE TESTING PROGRAM (TAC NOS. MB3685 and MB3686)

REF: 1) TXU Generation Company LP Letter logged TXX-01187, from C. L. Terry to the NRC dated December 26, 2001

Gentlemen:

Pursuant to 10CFR50.90, TXU Generation Company LP requested, via Reference 1, an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating a change into the CPSES Unit 1 and 2 Technical Specifications. The change request applies to both units.

The proposed change, as submitted by Reference 1, will revise TS 5.5.16 entitled Containment Leakage Rate Testing Program. This request proposes a one-time extension of the ten-year period of the performance-based leakage rate testing program for Type A tests as prescribed by NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR Part 50, Appendix J," and applied by 10CFR50, Appendix J, Option B. The ten-year interval between integrated leakage rate tests is to be extended to 15 years from the previous integrated leakage rate tests, which were completed on December 7, 1993 (Unit 1) and December 1, 1997 (Unit 2). The change reflects a one-time deferral of the next Type A Containment Integrated Leak Rate Test (ILRT) to no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2). This proposed change is based on and has been evaluated using the "risk informed" guidance in Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

D029
A017

As a result of subsequent conversations with your NRC staff (D. H. Jaffe), it was agreed that certain pages of the non-proprietary Enclosure 3 to Reference 1 would be re-issued. The affected pages were reviewed and proprietary information removed. Page "B.1" of Enclosure 3 to Reference 1 should be removed and replaced with the pages from Enclosure 1 of this letter. This letter is an administrative update to a previously submitted non-proprietary document.

The information in this supplement does not affect the proposed Technical Specification changes, the safety analysis of those changes, or the determination that the proposed changes do not involve a significant hazard consideration (provided by Attachments 1, 2 and 3 of Reference 1).

This communication contains no new or revised commitments.

Should you have any questions, please contact Mr. Carl B. Corbin at (254) 897-0121.

I state under penalty of perjury that the foregoing is true and correct.

Executed on February 4, 2002.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

C. L. Terry
Senior Vice President and Principal Nuclear Officer

By: Roger D. Walker

Roger D. Walker
Regulatory Affairs Manager

CBC/cbc

Enclosure 1. Comanche Peak Steam Electric Station Probabilistic Safety Assessment, Evaluation of Risk Significance of ILRT Extension (Non-proprietary replacement pages)

c - E. W. Merschoff, Region IV
D. N. Graves, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

Mr. Authur C. Tate
Bureau of Radiation Control
Texas Department of Public Health
1100 West 49th Street
Austin, Texas 78704

ENCLOSURE 1 to TXX-02023

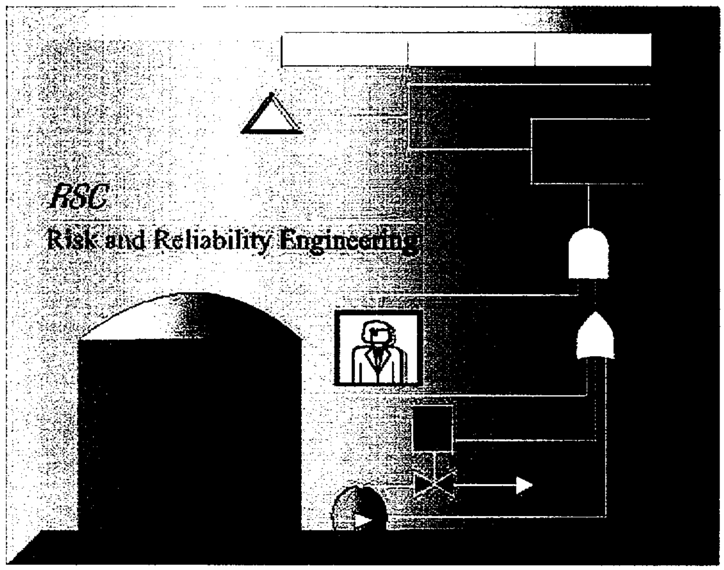
Comanche Peak Steam Electric Station Probabilistic Safety Assessment, Evaluation of Risk Significance of ILRT Extension

[Non-Proprietary Replacement Pages for Enclosure 3 to TXX-01187]

Note: Page "B.1" of Enclosure 3 TXX-01187 should be removed and replaced with the pages from this Enclosure (excluding this page, 16 pages total).

Appendix B

Surrogate Person-Rem Methodology
(RSC 01-44)



Surrogate Level 3 Evaluation Methodology

Revision 0

August 2001

Principal Analyst

Ricky Summitt

NON-PROPRIETARY DOCUMENT
This document has been reviewed and proprietary information removed. It may be freely distributed as a complete document only.

RSC Risk and Reliability Engineering

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RSC Document Configuration Control Form
FORM NO.: RSC-RPT-STD99-04, Rev. 4

Report Number: RSC 01-44NP

Title: Surrogate Level 3 Evaluation Methodology

Revision: Revision 0

Author: Ricky Summitt

Date Completed:	August 6, 2001
Location on Server of Report Files:	//rscsvr1/rsc_internal_reports
Location of Report and Files RSC Site:	NA
Date of Record for all Models and Analysis	August 6, 2001
Software and Version Used: (bold all that apply)	Word (doc) Version 95, Version 97, **Version 2000** Excel (xls): Version 95, Version 97, **Version 2000** Access (mdb): Version 97, Version 2000 Designer (ds4/f): Version 7 CAFTA: Version 3.2b ETA Version 3.2b MAAP BWR Version 3.0B R9, R11, Version 4.0 MAAP PWR Version 18, 19, 20, Version 4.0 RSC Software: PRAMS, SIP, TIFA, BAYESUPDATE

Report Review and Resolution Form
FORM NO.: RSC-RPT-RVR00-02Rev. 2

Preparer:	Ricky Summitt
RSC Reviewer: R. Summitt	Date: August 6, 2001
RSC Approver: R. Summitt	Date: August 6, 2001

Abstract (brief statement of purpose): Document methodology for converting radionuclide release fractions to dose. NOTE: Document grandfathered and does not require independent review since prior client review.

Documentation Retrieval Information:

Keywords: Level 2 Analysis Other Calculation MAAP Analysis

☒ Amends / ☐ Superceeds / ☐ Supplements RSC Document(s): PSA Paper (Reference 1).

Verification and Review Method:

☐ Detailed Review ☐ Alternative Calculation
☐ Qualification Testing ☒ Other (specify: Grandfathered)

General Documentation Requirements	Acceptable	Reviewer Comments
Introduction – provides summary of purpose, scope, and principle tasks required to meet objective	☐	
Methodology – description of process and supporting methodology that is sufficient to understand approach and to support peer review	☐	
Analysis and Results – detailed documentation of the implementation of the methodology and task steps that may be supported by report appendices and includes intermediate and final results	☐	
Conclusions and Recommendations – concise presentation of results of the analysis that answers the objectives of the study and should include any important assumptions and/or findings	☐	NA

Editorial Review:

☒ Spell Checked ☒ Grammer Checked
☒ Tables and Figures Checked ☒ Sections Checked

Sufficient References to Reproduce Results: Yes

Resolved all Comments: NA Incorporated Resolutions From Review: NA

Reviewer Comment	Resolution of Comment
1. NA	

Editorial or illustrative comments are attached to this review sheet to complete the review package.

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1.0 INTRODUCTION

The current industry emphasis is on applying the PSA to assist in plant operational decision-making. Most of the IPE submittals stop at the frequency of containment release and do not address offsite consequences. Since public safety is a primary consideration, it is important to have a tool that provides insights into how potential changes will impact public health risk.

Although a primary measure currently being proposed examines changes in the large early release fraction (LERF), the total effect should also be considered when evaluating changes.

The total whole body person-rem released is one measure to address the change in public health risk due to a proposed change to plant configuration. This quantity is considered one possible measure of merit and is traditionally calculated for the Level 3 PSA.

Given that most PSAs stop at containment release, additional effort is needed. To generate the person-rem release in order to expand the evaluation it is necessary to develop a model for extrapolating the existing information in the PSA to person-rem.

One approach to accomplish this task is to expand the existing PSA into a Level 3 PSA. This requires information on meteorological conditions, population densities, and evacuation planning. This information is then input into an offsite analysis code and results generated. The effort required to develop this detailed model may not be necessary for most cases.

A surrogate model can be used to estimate the change in whole body person-rem based on existing analyses[1]. The process used to develop the model is present in this report.

2.0 METHODOLOGY

The basis for the surrogate model is the development of a relationship between the radionuclide release fractions and the predicted whole body person-rem. To make the model useful, this relationship is developed at a release category level and in terms of a minimal set of radionuclide release fractions that, based on prior studies, can be shown to control the various aspects of offsite doses. This is accomplished by examining several prior studies that included measures of offsite consequences.

3.0 DEVELOPMENT OF RADIONUCLIDE RELEASE TO PERSON-REM RELATIONSHIP

The understanding that the dose values must be considered in terms of the "fence post" dose is key to the model development. In other words, the dose that the envelop around the plant would receive. This allows the results to be independent of evacuation and meteorological considerations. The result may be somewhat conservative, but it provides a measure that can be applied across plant sites uniformly.

3.1 DATA ASSESSMENT EXISTING

The results of the Level 2 IPE assessment are typically provided in terms of release category frequencies and radionuclide release fractions. Therefore, any method must utilize these two

characteristics form the basis for estimating the offsite consequence from release sequences to be useful.

To determine this relationship, available published and unpublished Level 3 PSAs were reviewed to determine a range of release fractions and corresponding doses. The release fractions identified in these PSAs for the following radionuclides: noble gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium and barium. The relative release fractions for each were collected as identified in the PSAs.

These radionuclides are most reported in the literature and provide the majority of offsite dose. The release fractions for each of the release categories is cataloged (each release category is defined as a *case*) along with the associated whole body person-rem. Figures 1 through 4 graphically presents the results for four PSAs as examples of this effort.

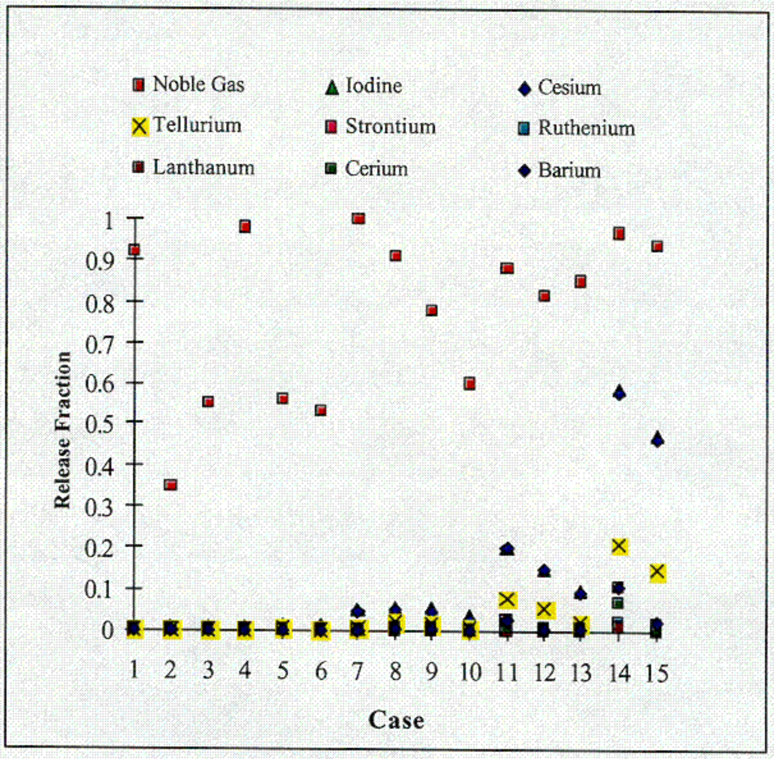

Figure 1 Sequoyah Release Fraction Cases (Reference 2)

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[• • •]

Figure 2 Unpublished PWR Release Fraction Cases (Reference 3)

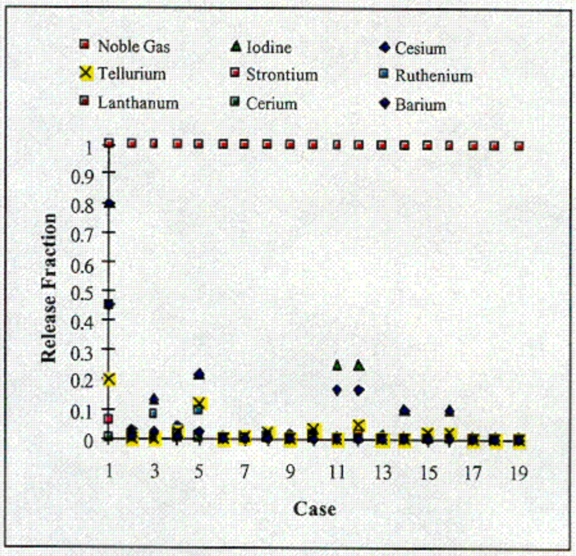

Figure 3 Oconee IPE Release Fraction Cases (Reference 4)

C02

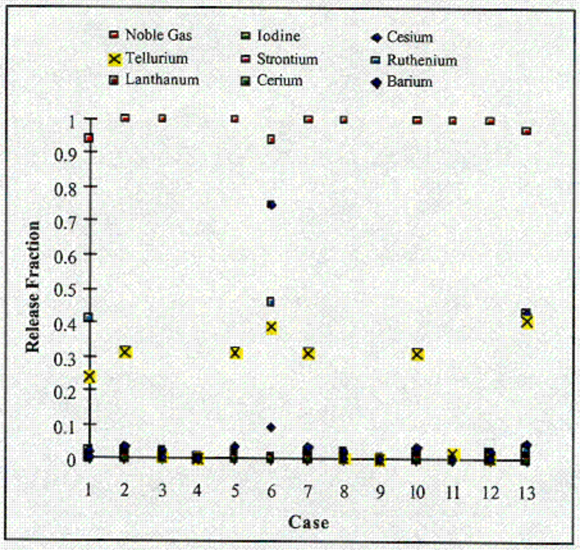

Figure 4 Seabrook Release Fraction Cases (Reference 5)

3.2 DATA INTERPRETATION

From these studies, a total of 56 unique release categories, defining radionuclide fractions and person-rem were plotted on a normalized plot to determine the type of relationship that existed between dose and release fractions. Five of the more important radionuclides were used to develop the release fraction value. These five radionuclides, noble gases, [···],[···],[···], and [···], are all considered important contributors to offsite dose.

Noble gas releases were chosen to represent the "baseline" dose. Most studies indicate that if a release occurs, the vast majority of noble gases will be released. The others were chosen based on their relatively important biological effects and tend to be significant release contributors. Figure 5 shows how the dose essentially maps the release fraction.

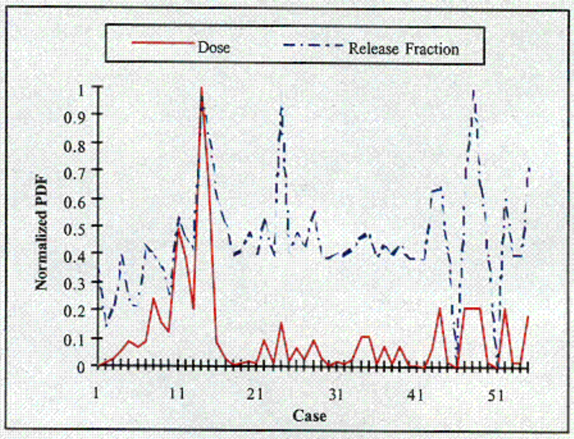

Figure 5 Relationship between Release Fraction and Dose

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Although a clear linear relationship does not exist between the two functions, it is clear that a trend is found between the fraction released and the resulting dose. This is hardly a revelation since the dose exposure is a function of the radionuclides released. The simplicity of the relationship, [···], is somewhat of a surprise. Given this relationship, a set of 56 [···] equations was developed. For each case, the equation took the form:

[···]

where: D_i=dose for case i

Xn_i = the release fraction for the key radionuclide n and case i

A,B,C,D, and E are constants.

These equations were setup as a series of simultaneous equations and the constants varied until an optimal solution to all equalities was determined. The correlation was obtained by matching the values generated by the equation to the whole body dose reported in the literature. Figure 6 presents the correlation for the 56 cases obtained for the final solution.

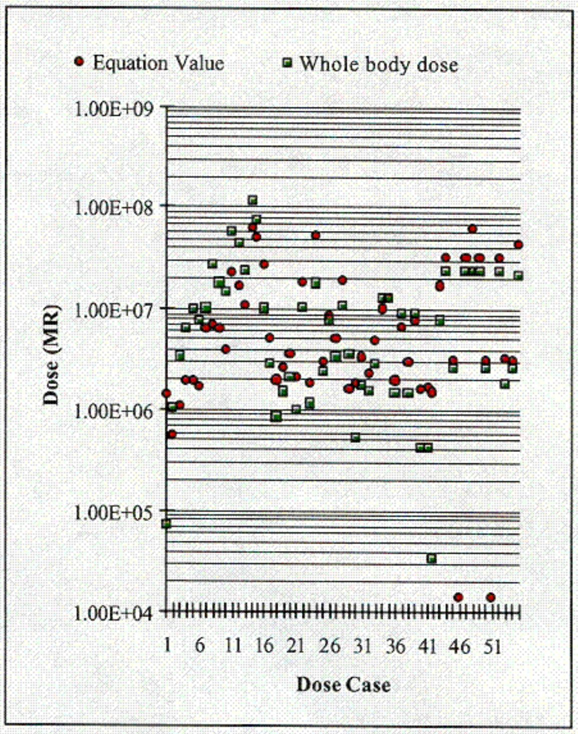

Figure 6 Comparison of Equation Results and Reported Dose Values

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The factors used to serve as constants that provide the best solution are presented in Table 1.

Table 1
Release Split Fraction to Dose Conversion Factors

Constant	Radionuclide Group	Value
A	Noble gases	[···]
B	[···]	[···]
C	[···]	[···]
D	[···]	[···]
E	[···]	[···]

3.4 APPLICATION WITH MAAP

The MAAP code provides radionuclide release fractions for significant radionuclides given a failure of containment. The release fractions can be used along with the method presented in this document to estimate the person-rem release.

In order to perform the calculation it is necessary to define what radionculide categories, as defined by MAAP, are needed. Table 2 lists the radionuclide categories utilized and how these radionuclides are mapped to the variables in the methodology.

Table 2
Mapping of Method Variables to MAAP Output Variables

Equation Variable	MAAP Output Variables
X1	Noble gas
X2	[···]
X3	[···]
X4	[···]
X5	[···]

Several of the surveyed PSAs utilized MAAP results to define the release category source term and the correlation has shown to be applicable if these MAAP variables are utilized.

3.5 QUALITATIVE UNCERTAINTY ASSESSMENT

The objective of this activity is to develop a realistic tool for estimation of person-rem. The process must not introduce excessive or unpredictable uncertainty. Two aspects of uncertainty that impact the analysis are the uncertainty in the generated magnitude and the consistency of the overall predictions.

3.5.1 Qualitative Evaluation of Predictive Dose

In addition to choosing the best fit for the 56 cases, the variation of the result for each unique case was examined. Figure 7 plots the variation from the reported value for each of cases. The range represents a deviation of a factor of two (2) in either direction.

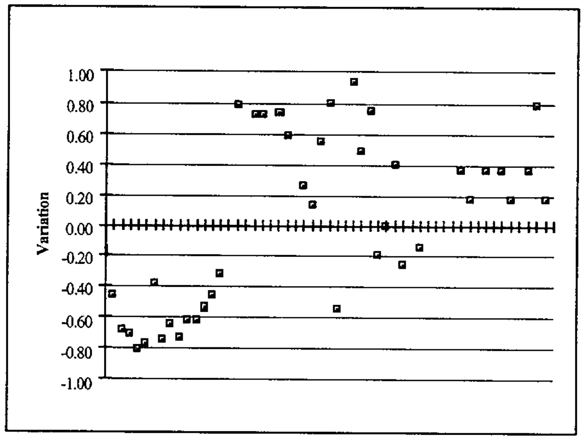

Figure 7 Variation of Equation to Reported Dose

As shown, most calculated values do not vary from the reported value by more than 50%. Given that the most likely use of this evaluation is to perform an assessment of relative change and that large uncertainties are already present in the PSA, errors of this magnitude (less than a factor of 2) are not significant.

The equation, however, was found to significantly over predict dose for cases involving intact containment leakage rates. In these cases, the offsite dose was less than 1.0E+5 person-rem and the variation approached a factor of 50. Thus, the equation may not be appropriate for intact containment cases. The cause of this error is the noble gas contribution. A basic assumption for impaired containment cases is that essentially 100% of noble gases are released such that the noble gas release is essentially a baseline dose as stated earlier. This is not the case for intact containments and the constant chosen for the noble gas contribution is significantly overestimated. This limitation, however, does not affect the use of this model since any assessment would be based on results for impaired containment events. Existing licensing basis analyses can cover intact containment doses and it is this data that is the support for the intact containment release category.

3.5.2 Results Predictability

To have confidence in the method it is necessary for the analysis to be internally consistent. This does not preclude generating conservative or non-conservative results. It does require that the results generated are not bimodal resulting in significant differences in the trend of the results. For example, if one release category is underestimated and another overestimated the importance of the two release categories will be incorrect. If both are slightly overestimated the relative importance will be maintained.

An evaluation of the results (see Figures 6 and 7) indicates that the model consisting estimates a value slightly greater than the reference value. For intact containment cases, however, this was not the case. The value was significantly overestimated and again this supports not using this approach for intact containment cases. Figure 7 also shows several cases when the values were slightly under predicted. This was a single plant with an older evaluation of source term not representative of the current state of knowledge and the underestimation is appropriate and more representative of expected source term. Again the analysis is internally consistent. The method is consistent to provide predictable results and the uncertainty from this aspect is small.

4.0 SUMMARY AND CONCLUSIONS

A simplified model for addressing offsite risk is possible using existing PSA information and can be based on relatively few radionuclides. The development of this model can provide a useful tool to evaluate potential plant configuration changes and improvements.

The use of this model to calculate the impact of proposed changes can be used to assess the impact of procedural changes, operating status, or other modifications on a relative change in whole body person-rem.

It is important to mention that person-rem is only one of the factors that should be considered and that it is not usually the most restrictive when evaluating total risk. The lost plant investment and replacement power costs must also be considered internally in the decision process. The use of a health risk measure such as person-rem, however, does provide a type of regulatory perspective on potential changes in plant status or configuration.

5.0 REFERENCES

1. Summitt, R., *Development of a Surrogate Risk Measure for Risk Benefit Assessment*, PSA 96, September 29-October 3, 1996.

2. Benjamin et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Sequoyah Power Station, Unit 1, United State Regulatory Commission, NUREG/CR-4551, Vol. 2, in press.

3. Unpublished PSA.

4. W. Sugnet et al., Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3, The Nuclear Safety Analysis Center and Duke Power Company, NSAC-60, June 1984.

5. Garrick et al., Seabrook Station Probabilistic Safety Assessment, Pickard, Lowe, and Garrick, Inc., PLG-0300, December 1983.