Dave Morey
Vice President
Farley Project

Southern Nuclear
Operating Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.5131

SOUTHERN COMPANY
Energy to Serve Your World℠

April 4, 2002

Docket Nos.: 50-348
50-364

NEL-02-0001

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50, Southern Nuclear Operating Company (SNC) is proposing a change to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 and Unit 2 Technical Specifications (TS). This proposed change will revise TS section 5.5.17, "Containment Leakage Rate Testing Program," to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The ten (10) year interval between integrated leakage rate tests is to be extended to fifteen (15) years from the previous integrated leakage rate tests, which were completed in March 1994 (Unit 1) and March 1995 (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." The "Risk Assessment for Joseph M. Farley Nuclear Plant Regarding (Type A) Extension Request" is provided as an attachment to this letter.

Enclosure 1 provides a description of the proposed change and an explanation of the basis for the change. Enclosure 2 details the basis for SNC's determination that the proposed change does not involve a significant hazards consideration. Enclosure 3 provides page change instructions for incorporating the proposed change along with the revised Technical Specification page and the corresponding marked-up page.

Southern Nuclear Operating Company requests the proposed amendment be approved by November 2002 to support the planning activities for the Unit 1 outage scheduled in March 2003.

A017

A similar request was approved for Indian Point 3 in a letter dated April 17, 2001, Crystal River 3 in a letter dated August 30, 2001, and Peach Bottom 3 in a letter dated October 4, 2001.

This letter contains no new commitments.

A copy of the proposed changes has been sent to Dr. D. E. Williamson, the Alabama State Designee, in accordance with 10 CFR 50.91(b)(1).

Mr. D. N. Morey states that he is a Vice President of Southern Nuclear Operating Company and is authorized to execute this oath on behalf of Southern Nuclear Operating Company and that, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

DN Morey

Dave Morey

Sworn to and subscribed before me this 4th *day of* April *2002*

Gloria H. Buie

Notary Public

My Commission Expires: June 7, 2005

CHM: ILRT Ext to 15.doc

Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. FNP Technical Specification Changed Page List, Marked-up Page and Typed Page

Attachment: Risk Assessment for Joseph M. Farley Nuclear Plant Regarding (Type A) Extension Request

cc: Southern Nuclear Operating Company
Mr. D. E. Grissette, General Manager – Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. F. Rinaldi, Licensing Project Manager – Farley

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Joseph M. Farley Nuclear Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Enclosure 1

Basis for Change Request

Joseph M. Farley Nuclear Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Enclosure 1

Basis for Change Request

Proposed Change

Southern Nuclear Operating Company (SNC) is proposing a change to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 and Unit 2 Technical Specifications (TS). This proposed change will revise TS section 5.5.17, "Containment Leakage Rate Testing Program," to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The ten (10) year interval between integrated leakage rate tests is to be extended to fifteen (15) years from the previous integrated leakage rate tests, which were completed in March 1994 (Unit 1) and March 1995 (Unit 2).

The proposed change involves a one time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J." The current ten (10) year ILRT for FNP is due in March 2004 (Unit 1) and March 2005 (Unit 2), which would require the test to be performed during Refueling Outage Unit 1-R18 (March 2003) and Unit 2-R16 (March 2004). The proposed exception would allow the next ILRTs for FNP to be performed within fifteen (15) years (Unit 1 - March 2009 and Unit 2 – March 2010) from the last ILRTs as opposed to the current ten (10) year frequency.

The proposed change would revise Section 5.5.17, "Containment Leakage Rate Testing Program," of the FNP Technical Specifications as follows:

> A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, **as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":**
>
> **Section 9.2.3: The next Type A test, after the March 1994 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years. This is a one time exception.**

This one-time exception will result in the following:

- For Unit 1, the Type A Containment ILRT test will be performed during Refueling Outage Unit 1-R22, currently scheduled for March 2009.
- For Unit 2, the Type A Containment ILRT test will be performed during Refueling Outage Unit 2-R20, currently scheduled for March 2010.

- A substantial cost savings will be realized and unnecessary personnel radiation exposure will be avoided by deferring the Type A test for an additional five (5) years. Cost savings have been estimated for each outage at approximately $1.95 million, which includes labor, equipment and critical path outage time needed to perform the test. Personnel radiation exposure reduction is estimated at 750 mrem.

Basis for Proposed Change

a. <u>10 CFR 50, Appendix J, Option B</u>

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in Technical Specifications. The limitation on containment leakage provides assurance that the containment will perform its design function following plant design basis accidents.

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements" or Option B, "Performance-Based Requirements." Amendment 122 was issued for FNP Unit 1 (dated September 3, 1996) and Amendment 114 was issued for FNP Unit 2 (dated September 3, 1996) to permit implementation of 10 CFR 50, Appendix J, Option B. These Amendments revised Technical Specifications to require Type A, B, and C testing in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." RG 1.163 specifies a method acceptable to the NRC for complying with 10 CFR 50, Appendix J, Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to several regulatory positions in the guide.

Exceptions to the requirements of RG 1.163 are permitted by 10 CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10 CFR 50, Appendix J, Option B.

Adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, B, and C containment leakage tests must be performed. Under the performance based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews "as found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowed frequency for Type A testing, as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493. The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident risks on containment leak-tightness for five reactor/containment types including a Westinghouse designed pressurized water reactor in a large, dry containment building, (FNP Unit 1 and 2 are large, dry containment buildings). NUREG-1493 made the following observations with regard to decreasing the test frequency.

- Reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one per twenty (20) years was found to lead to imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

NEI 94-01 requires that Type A testing be performed at least once per ten (10) years based upon an acceptable performance history. Acceptable performance history is defined as two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a. Based upon the acceptable ILRTs for Unit 1 (May 1991 and March 1994,) and for Unit 2 (December 1990 and March 1995) the current test interval for FNP is once every ten (10) years, with the next test due to be performed by March 2004 on Unit 1 and March 2005 on Unit 2.

b. FNP Integrated Leak Rate Test History

Type A testing is performed to verify the integrity of the containment structure in its Loss of Coolant Accident (LOCA) configuration. Industry test experience has demonstrated that Type B and C testing detect a large percentage of containment leakage and that the percentage of containment leakage that is detected only by integrated containment leakage testing is very small.

FNP Unit 1 has undergone 5 operational Type A tests and Unit 2 has undergone 4 operational Type A tests, in addition to the pre-operational Type A test performed on each unit. The results of these tests demonstrate that the FNP containment structures, for Unit 1 and Unit 2, remain essentially leak-tight barriers and represent minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493. As specified in FNP Technical Specifications Section 5.5.17, the maximum allowable containment leakage rate L_a, at P_a, is 0.15% of containment air weight per day. The FNP ILRT results are provided below.

Unit 1

Pre-Operational completed February 1977
Mass Point Analysis (95% upper confidence limit) = 0.085 (weight % per day)

1st Periodic January 1981
Mass Point Analysis (95% upper confidence limit) = 0.054 (weight % per day)

2nd Periodic April 1984
Mass Point Analysis (95% upper confidence limit) = 0.088 (weight % per day)

3rd Periodic November 1986
Mass Point Analysis (95% upper confidence limit) = 0.042 (weight % per day)

4th Periodic May 1991
Mass Point Analysis (95% upper confidence limit) = 0.055 (weight % per day)

5th Periodic March 1994
Mass Point Analysis (95% upper confidence limit) = 0.048 (weight % per day)

Unit 2

Pre-Operational completed June 1980
Mass Point Analysis (95% upper confidence limit) = 0.061 (weight % per day)

1st Periodic March 1985
Mass Point Analysis (95% upper confidence limit) = 0.053 (weight % per day)

2nd Periodic November 1987
Mass Point Analysis (95% upper confidence limit) = 0.064 (weight % per day)

3rd Periodic December 1990
Mass Point Analysis (95% upper confidence limit) = 0.048 (weight % per day)

4th Periodic March 1995
Total Time Analysis (95% upper confidence limit) = 0.111 (weight % per day)

c. Containment Inspections

Containment Leak tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), section XI. More specifically, subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments in light water cooled plants. Furthermore, NRC regulations, 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas on the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak tight integrity of containment penetration bellows, airlocks, seals and gaskets are not affected by the change to the Type A test frequency. Likewise the Appendix J, Type C local leak tests, which are performed to verify the leak tight integrity of containment isolation valves, are not affected by the change to the Type A test frequency.

d. Risk Assessment

Attached is a detailed performance based, risk informed assessment, "Risk Assessment for Joseph M. Farley Nuclear Plant ILRT (Type A) Extension Request," to support this request.

e. Similar Requests

This request is similar to the requests for change of the Indian Point 3 ILRT frequency that was approved by the NRC on April 17, 2001, Crystal River 3 approved in an NRC letter dated August 30, 2001, and Peach Bottom 3 approved in an NRC letter dated October 4, 2001. The PRA has been enhanced with the knowledge gained from the NRC's evaluation of the recent Crystal River Unit 3 submittal.

f. Conclusion

Based on the attached risk assessment results, the containment leak rate test results and containment inspection results, the requested change is concluded to be acceptable.

Joseph M. Farley Nuclear Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Enclosure 2

10 CFR 50.92 Evaluation

Joseph M. Farley Nuclear Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Enclosure 2

10 CFR 50.92 Evaluation

In 10 CFR 50.92 (c), the NRC provides the following standards to be used in determining the existence of a significant hazards consideration:

> ... a proposed amendment to an operating license for a facility licensed under §50.21(b) or §50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

Southern Nuclear Operating Company has reviewed the proposed license amendment request and determined its adoption does not involve a significant hazards consideration based on the following discussion.

Basis for no significant hazards consideration determination

1. *The proposed Technical Specifications change does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The proposed revision to Technical Specifications 5.5.17, "Containment Leakage Rate Testing Program," involves a one time extension to the current interval for Type A containment leak testing. The current test interval of ten (10) years would be extended on a one time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specifications change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

   The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG-1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. FNP test history supports this conclusion. NUREG-1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanism which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the containment coatings program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specifications change does not involve a significant increase in the consequences of an accident previously evaluated.

2. *The proposed Technical Specifications change does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

   The proposed revision to Technical Specifications involves a one time extension to the current interval for Type A containment leak testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specifications change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specifications change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *The proposed Technical Specifications change does not involve a significant reduction in a margin of safety.*

   The proposed revision to Technical Specifications involves a one time extension to the current interval for Type A containment leak testing. The proposed Technical Specifications change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

   FNP and industry experience strongly support the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the Coatings Program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specifications change does not involve a significant reduction in a margin of safety.

Environmental Impact

The proposed Technical Specifications changes were reviewed against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant increase in the amounts of effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposures. Based on the foregoing, Southern Nuclear Operating Company concludes that the proposed Technical Specifications change meets the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

Conclusion

SNC has concluded that the proposed change to the FNP Technical Specifications does not involve a Significant Hazards Consideration.

Joseph M. Farley Nuclear Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Enclosure 3

FNP Technical Specification Changed Page List

FNP Technical Specification Marked-up Page

FNP Technical Specification Typed Page

Joseph M. Farley Nuclear Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Changed Page List

Changed Page	Revision Instruction
5.5-17	Replace

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Main Steamline Inspection Program

The three main steamlines from the rigid anchor points of the containment penetrations downstream to and including the main steam header shall be inspected. The extent of the inservice examinations completed during each inspection interval (IWA 2400, ASME Code, 1974 Edition, Section XI) shall provide 100 percent volumetric examination of circumferential and longitudinal pipe welds to the extent practical. The areas subject to examination are those defined in accordance with examination category C-G for Class 2 piping welds in Table IWC-2520.

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

Section 9.2.3: The next Type A test, after the March 1994 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years. This is a one time exception.

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Main Steamline Inspection Program

The three main steamlines from the rigid anchor points of the containment penetrations downstream to and including the main steam header shall be inspected. The extent of the inservice examinations completed during each inspection interval (IWA 2400, ASME Code, 1974 Edition, Section XI) shall provide 100 percent volumetric examination of circumferential and longitudinal pipe welds to the extent practical. The areas subject to examination are those defined in accordance with examination category C-G for Class 2 piping welds in Table IWC-2520.

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

Section 9.2.3: The next Type A test, after the March 1994 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years. This is a one time exception.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

(continued)

Joseph M. Farley Nuclear Plant
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Attachment

Risk Assessment for Joseph M. Farley Nuclear Plant
Regarding (Type A) Extension Request

RISK ASSESSMENT FOR

Joseph M. Farley Nuclear Plant

REGARDING

ILRT (TYPE A) EXTENSION REQUEST

Prepared for:

Southern Nuclear Operating Co.

Prepared by:

ERIN *ENGINEERING AND RESEARCH, INC*
1210 Ward Avenue, Suite 100
West Chester, PA 19830

March 2002

Joseph M. Farley Nuclear Plant

RISK ASSESSMENT FOR Joseph M. Farley Nuclear Plant REGARDING ILRT (TYPE A) EXTENSION REQUEST

Prepared by: Donald E. Vanover Date: 3/8/02

Reviewed by: Charles A. Menges Date: 3/10/02

Approved by: [illegible] Date: 3/10/02

Accepted by: William E. [illegible] Date: 3/08/02

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date

TABLE OF CONTENTS

Section 1
PURPOSE OF ANALYSIS

1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Joseph M. Farley Nuclear Plant. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals from November 2001 [14], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide 1.174 [3].

1.1 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than normal containment leakage of 1.0La.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

Previously, the NRC published a report, Performance Based Leak Test Program, NUREG-1493 [4], which analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures for Farley.

NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals dated November 2001 builds on the EPRI Risk Assessment methodology, EPRI TR-104285 (Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals). This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency.

1.2 CRITERIA

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 10^{-6} per reactor year. RG 1.174 discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability that helps to ensure that the defense-in-depth philosophy is maintained will also be calculated.

In addition, the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in their parameters. (No criteria have been established for this parameter change.)

Section 2
METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in NEI Interim Guidance [14], EPRI TR-104285 [2] and NUREG-1493 [4]. The analysis uses the current Farley PSA model that includes the results from the Farley Level 2 analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no or negligible release).

The four general steps of this risk assessment are as follows:

1) Quantify the baseline risk and sensitivity cases in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.

2) Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.

3) Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.

4) Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with Regulatory Guide 1.174 [3] and compare with the acceptance guidelines of RG 1.174.

This approach is based on the information and an approach contained in the previously mentioned studies and further is consistent with the following:

- Consistent with the other industry containment leak risk assessments, the Farley assessment uses population dose as one of the risk measures. The other risk measures used in the Farley assessment are Large Early Release Frequency (LERF) and Conditional Containment Failure Probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.

- Consistent with the approach used in the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [14], the FNP evaluation uses similar ground rules and methods to calculate changes in risk metrics.

Section 3

GROUND RULES

The following ground rules are used in the analysis:

- The Farley Level 1 and Level 2 internal events PSA models provide representative results for the analysis.
- It is appropriate to use the Farley internal events PSA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations.
- Dose results for the containment failures modeled in the PSA can be characterized by information provided in NUREG/CR-4551 [9]. They are estimated by scaling the NUREG/CR-4551 results by population differences for FNP compared to the NUREG/CR-4551 reference plant.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is 1 L_a. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10 L_a. based on the previously approved methodology [6, 7].
- The representative containment leakage for Class 3b sequences is 35 L_a. based on the previously approved methodology [6, 7].
- For Unit 1, Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [6, 7]. The Class 3b category increase is used as a surrogate for LERF in this application even though the releases associated with a 35La release would not necessarily be consistent with a "Large" release for Farley. (See, for example, the calculated population dose results for EPRI Class 3b in Table 5-3 of 5.18E3 person-rem compared to the 2.84E5 person-

rem associated with EPRI Class 8 for Containment Bypass scenarios.)

- For Unit 2, however, the over-conservatism is accounted for in the analysis by obtaining a first order approximation of the percentage of the EPRI Class 3b scenarios that would more realistically constitute LERF for Farley. This must be kept in mind when comparing the calculated "LERF" changes per the NEI methodology with the acceptance guidelines from Reg. Guide 1.174.

- The impact on population doses from Containment Bypass scenarios is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the Containment Bypass contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this assumption.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

Section 4
INPUTS

This section summarizes the general resources available as input (Section 4.1) and the plant specific resources required (Section 4.2).

4.1 GENERAL RESOURCES AVAILABLE

Various industry studies on containment leakage risk assessment are briefly summarized here:

1) NUREG/CR-3539 [10]
2) NUREG/CR-4220 [11]
3) NUREG-1273 [5]
4) NUREG/CR-4330 [12]
5) EPRI TR-105189 [8]
6) NUREG-1493 [4]
7) EPRI TR-104285 [2]
8) NUREG-1150 [13] and NUREG/CR-4551 [9]
9) NEI Interim Guidance from November 2001 [14]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. The eight studies provide an ex-plant consequence analysis for a 50-mile radius surrounding a plant that is used as the bases for the consequence analysis of the ILRT interval extension for FNP. Finally, the ninth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [18] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage.

NUREG-1273 [5]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [12]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [8]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk. The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals.

NUREG-1493 [4]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk

- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage frequencies into eight (8) classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

"These study results show that the proposed CLRT [containment leak rate tests] *frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year . . ."*

NUREG-1150 [13] and NUREG/CR 4551 [9]

NUREG-1150 and the technical basis, NUREG/CR-4551 [9], provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Tech Spec leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the FNP Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent FNP. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [14]

The guidance provided in this document builds on the EPRI Risk Impact Assessment methodology [2] and the NRC Performance-Based Containment Leakage Test Program [4], and considers approaches utilized in various submittals, including Indian Point 3 (and associated NRC SER) and Crystal River.

The approach included in this guidance document is used in the FNP assessment to determine the estimated increase in risk associated with the ILRT extension. This document includes the bases for the values assigned in determining the probability of leakage for the EPRI Class 3a and 3b scenarios as used in this analysis as described in Section 5.

4.2 PLANT SPECIFIC INPUTS

The information used to perform the FNP ILRT Extension Risk Assessment includes the following:

- Level 1 Model results
- Level 2 Model results
- Release Category definitions used in the Level 2 Model
- Population Dose calculations by release category
- ILRT results to demonstrate adequacy of the administrative and hardware issues.[1]

Level 1 Model

The Level 1 PSA model that is used for FNP is characteristic of the as-built plant. The current Level 1 model is developed in CAFTA, and was quantified with the total Core Damage Frequency (CDF) = 3.85E-5/yr for Unit 1 and 5.81E-5/yr for Unit 2.

Level 2 Model

The Level 2 Model that is used for FNP was developed to calculate the LERF contribution as well as the other release categories evaluated in the model. The Level 2 model was also quantified using the CAFTA model. The total Large Early Release Frequency (LERF) which corresponds to Farley release categories D,G, and T in Table 4.2-1 was found to be 4.4E-7/yr for Unit 1 and 4.7E-7 for Unit 2 at a truncation of 5E-13/yr. Table 4.2-1 summarizes the pertinent FNP results in terms of release category.

(1) The two most recent Type A tests at FNP Unit 1 and FNP Unit 2 have been successful, so the current Type A test interval requirement is 10 years. In fact, the last 3 ILRT results at the FNP Units 1 and 2 have been successful [19].

Table 4.2-1
FNP Level 2 PSA Model Release Categories and Frequencies

Release Category	Definition	Unit 1 Frequency/yr	Unit 2 Frequency/yr
A	No containment failure within 48-hour mission time, but failure could eventually occur without further mitigating action; noble gases and less than 0.1% volatiles released	3.01E-5	4.46E-05
D	Containment bypassed with noble gases and up to 10% of the volatiles released	8.36E-8	8.65E-08
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment not isolated)	2.39E-8	5.03E-08
K	Late containment failure with noble gases and less than 0.1% volatiles released(containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful prior to core damage)	8.00E-6	1.30E-05
T	Containment bypassed with noble gases and more than 10% of the volatiles released.	3.34E-7	3.34E-07
	Total Release Category Frequency	3.85E-5	5.81E-05
	Core Damage Frequency	3.85E-5	5.81E-05

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for FNP. Each of the release categories from Table 4.2-1 was associated with an applicable Collapsed Accident Progression Bin (APB) from NUREG/CR-4551. The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 7 bins that are relevant to the analysis. The definitions of the 7 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table 4.2-2 for references purposes. Table 4.2-3 summarizes the calculated population dose for Surry associated with each APB from NUREG/CR-4551.

Table 4.2-2
Summary Accident Progression Bin (APB) Descriptions [9]

Summary APB Number	Description
1	CD, VB, Early CF, Alpha Mode Core damage occurs followed by a very energetic molten fuel-coolant interaction in the vessel; the vessel fails and generates a missile that fails the containment as well. Includes accidents that have an Alpha mode failure of the vessel and the containment except those follow Event V or an SGTR. It includes Alpha mode failures that follow isolation failures because the Alpha mode containment failure is of rupture size.
2	CD, VB, Early CF, RCS Pressure > 200psia Core Damage occurs followed by vessel breach. Implies Early CF with the RCS above 200 psia when the vessel fails. Early CF means at or before VB, so it includes isolation failures and seismic containment failures at the start of the accident as well as containment failure at VB. It does not include bins in which containment failure at VB follows Event V or an SGTR, or Alpha mode failures.
3	CD, VB, Early CF, RCS Pressure < 200 psia Core damage occurs followed by vessel breach. Implies Early CF with the RCS below psia when the containment fails. It does not include bins in which the containment failure at VB or an SGTR, or Alpha mode failures.
4	CD, VB, Late CF Core Damage occurs followed by vessel breach. Includes accidents in which the containment was not failed or bypassed before the onset of core-concrete interaction (CCI) and in which the vessel failed. The failure mechanisms are hydrogen combustion during CCI, Basemat Melt-Through (BMT) in several days, or eventual overpressure due to the failure to provide containment heat removal in the days following the accident.
5	CD, Bypass Core Damage occurs followed by vessel breach. Includes Event V and SGTRs no matter what happens to the containment after the start of the accident. It also includes SGTRs that do not result in VB.
6	CD, VB, No CF Core Damage occurs followed by vessel breach. Includes accidents not evaluated in one of the previous bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.
7	CD, No VB Core Damage occurs but is arrested in time to prevent vessel breach. Includes accident progressions that avoid vessel failures except those which bypass the containment. Most of the bins placed in this reduce bin have no containment failure as well as no VB. It also includes bins in which the containment is not isolated at the start of the accident and the core is brought to a safe stable state before the vessel fails.

Table 4.2-3
Calculation of Surry Population Dose Risk at 50 Miles

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR) [1]	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 5.55 person-rem/yr, mean) [2]	NUREG/CR-4551 Collapsed Bin Frequencies (per year) [3]	NUREG/CR-4551 Population Dose at 50 miles (Person-rem) [4]
1	0.029	0.158	1.23E-07	1.28E+06
2	0.019	0.106	1.64E-07	6.46E+05
3	0.002	0.013	2.012E-8	6.46E+05 [5]
4	0.216	1.199	2.42E-06	4.95E+05
5	0.732	4.060	5.00E-06	8.12E+05
6	0.001	0.006	1.42E-05	4.23E+02
7	0.002	0.011	1.91E-05	5.76E+02
Totals	**1.000**	**5.55**	**4.1E-05**	

(1) Mean Fractional Contribution to Risk calculated from the average of two samples delineated Table 5.1-3 of NUREG/CR-4551

(2) The total population dose risk at 50 miles from internal events in person-rem is provided as the average of two samples in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-3. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.

(5) Assumed population dose at 50 miles for Collapsed Bin # 3 equal to that of Collapsed Bin 2. Collapsed Bin Frequency # 3 was then back calculated using that figure. This does not influence the results of this evaluation since Bin 3 does not appear as part of the results for FNP.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for Farley Nuclear Plant if it is corrected for the population surrounding FNP. The total population within a 50 mile radius was supplied as part of the plant specific information provided by FNP [19]. The total population within 50 miles of FNP is 423,584.

This population data is compared to the population data that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that can be applied to the APBs to get dose estimates for FNP.

Total FNP Population$_{50miles}$ = 4.24E+05

Surrey Population from NUREG/CR-4551 = 1.23E+06

Population Dose Factor = 4.24E+05 / 1.23E+06 = 0.35

This population dose factor can then be applied to the APBs from NUREG/CR-4551. The difference in the doses at 50 miles is assumed to be in direct proportion to the difference in the population within 50 miles of each site. This does not take into account differences in meteorology data, detailed environmental factors, or detailed differences in containment designs, but does provide a first-order approximation for FNP of the population doses associated with each of the release categories from NUREG/CR-4551. This is considered adequate since the conclusions from this analysis will not be substantially affected by the actual dose values that are used.

Table 4.2-4 shows the results of applying the population dose factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for FNP.

Table 4.2-4
Calculation of FNP Population Dose Risk at 50 Miles

Accident Progression Bin (APB)	NUREG/CR-4551 Population Dose at 50 miles (Person-rem)	Bin Multiplier used to obtain FNP Population Dose	FNP Adjusted Population Dose at 50 miles (Person-rem)
1	1.28E+06	0.35	4.48E+05
2	6.46E+05	0.35	2.26E+05
3	6.46E+05	0.35	2.26E+05
4	4.95E+05	0.35	1.73E+05
5	8.12E+05	0.35	2.84E+05
6	4.23E+02	0.35	1.48E+02
7	5.76E+02	0.35	2.02E+02

Application of FNP PSA Model Results to NUREG/CR-4551 Level 3 Output

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the results of the FNP PSA Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to assign the FNP PSA Level 2 model results into a form which allowed for the scaling of the Level 3 results based on current Level 2 output. The assumptions used for these assignments are shown in Table 4.2-5.

Table 4.2-5
FNP Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins and EPRI / NEI Accident Classes

FNP Level 2 Release Category	Definition	NUREG/CR-4551 APB	EPRI/NEI Class
A	No containment failure within 48-hour mission time, but failure could eventually occur without further mitigating action; noble gases and less than 0.1% volatiles released	6	1
D	Containment bypassed with noble gases and up to 10% of the volatiles released	5	8
G	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released(containment not isolated)	2	2
K	Late containment failure with noble gases and less than 0.1% volatiles released(containment failure greater than 6 hours after vessel failure; containment not bypassed; isolation successful prior to core damage)	4	7
T	Containment bypassed with noble gases and more than 10% of the volatiles released.	5	8

Release Category Definitions

Table 4.2-6 defines the accident classes used in the ILRT extension evaluation consistent with the EPRI/NEI methodology [2].

Table 4.2-6
EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS

Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

These containment failure classifications are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

4.3 CONDITIONAL PROBABILITY OF ILRT FAILURE (SMALL AND LARGE)

The ILRT can detect a number of failures such as liner breach, failure of certain bellows arrangements, and failure of some sealing surfaces. The proposed ILRT test interval extension may influence the conditional probability associated with the ILRT failure. To ensure that this effort is properly accounted for, the Class 3 Accident Class as defined in Table 4.2-6 is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the NEI Guidance [14]. For Class 3a, the probability is based on the mean failure from the available data (i.e., 5 "Small" failures in 182 tests leads to a 5/182=0.027 mean value). For Class 3b, a non-informative prior distribution is assumed for no "Large" failures in 182 tests (i.e., 0.5/(182+1) = 0.0027).

In a follow on letter [15] to their ILRT guidance document [14], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain

conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.

The application of this additional guidance for Farley as detailed in section 5 involves the following:

1. The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large pre-existing containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Farley Level 2 PSA analysis.

2. A review of Class 1 accident sequences shows that several of these cases involve successful operation of containment sprays. It is assumed that, for calculation of the Class 3b and 3a frequencies, the fraction of the Class 1 CDF associated with successful operation of containment sprays can also be subtracted. A review of the Farley accident bins that contribute to Class 1 reveals that 65% of the accidents for Unit 1 include sprays and 43% for Unit 2. Table 4.3-1 provides a detailed breakdown of the sequences contributing to FNP release category A (NEI/EPRI Class 1). A review of the FNP release category K (NEI/EPRI Class 7) results in no credit for containment sprays. Sprays are not credited for any of the other release categories.

Table 4.3-1
FNP Level 2 Sequences Contributing to Release Category A

Bin #	Frequency		Spray [(1)] effective?
	Unit 1	Unit 2	
2	6.51E-6	6.61E-6	No
3	1.22E-7	1.23E-7	No
4	7.65E-6	7.09E-6	Yes
5	7.97E-6	7.95E-6	Yes
6	3.48R-6	1.79E-5	No
7	4.11E-6	4.10E-6	Yes
8	1.94E-7	8.03E-7	No
9	4.64E-9	5.04E-9	No
10	1.00E-7	1.00E-7	No
Total	3.01E-5	4.46E-5	

(1) Sprays are assumed to prevent a large release only if they are initiated prior to or immediately following vessel breach and continue to operate long term.

4.4 IMPACT OF EXTENSION ON LEAK DETECTION PROBABILITY

Consistent with the NEI Guidance [14], the change in probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yrs/2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yrs/ 2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. Correspondingly, an extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (e.g., the IP3 request for a one-time ILRT extension that was approved by the NRC on April 17,2000 (TAC No. MB0178 [7]) since it does not factor in the possibility that the failures could be detected by other tests (e.g., the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the factor increases attributable to the ILRT extension.

Section 5
RESULTS

The application of the approach based on NEI Interim Guidance [14], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6] have led to the following results.

The method chosen to display the results is according to the eight (8) accident classes consistent with these two reports. Table 5-1 lists these accident classes.

The analysis performed examined FNP specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the break down of the severe accidents contributing to risk were considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences). Consistent with the NEI Guidance, this Class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.
- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

Table 5-1
ACCIDENT CLASSES

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal –Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including very low and no release)

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 5-1.

Step 2 - Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes evaluated in EPRI TR-104285.

Step 3 - Evaluate risk impact of extending Type A test interval from 10 to 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

5.1 STEP 1 - QUANTIFY THE BASE-LINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the "Class 3" sequence depicted in EPRI TR-104285 [2]). The question on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two additional failure modes were considered in addition to large containment failure modes. These are Event CLASS-3A (small breach) and Event CLASS-3B (large breach).

After including the containment isolation failures, Class 2, and including the respective "large" and "small" liner breach leak rate probabilities, the eight severe accidents class frequencies were developed consistent with the definitions in Table 5-1 as described below.

<u>Class 1 Sequences</u>. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is initially determined from the Level 2 Release Category A minus the EPRI/NEI Class 3a and 3b frequency.

<u>Class 2 Sequences</u>. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the Release Category G frequency from table 4.2-1. The values of 2.39E-8 (Unit 1) and 5.03E-08 (Unit 2) were determined from the sum of all Level 2 end states involving containment isolation failure from the base model results.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small ($2L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.027 [see Section 4.3]

$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0027 [see Section 4.3]

As described in section 4.3, additional consideration is made to not apply these failure probabilities on those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions), or that would include containment spray operation such that a Large Release would be unlikely (i.e., 65% of the FNP Release Category A for Unit 1, and 43% of the FNP Release Category A for Unit 2).

For Unit 1:

CLASS_3A_FREQUENCY = 0.027 ∗ (CDF-Class 2-Class 8–0.65*Category A)
= 0.027 * (3.85eE-05 – 2.39E-08 – 4.20E-07 – 0.65 * 3.01E-05) = 4.99E-7/yr
CLASS_3B_FREQUENCY = 0.0027 ∗ (CDF-Class 2-Class 8–0.65*Category A)
=0.0027 * ((3.85eE-05 – 2.39E-08 – 4.20E-07 – 0.65 * 3.01E-05) = 4.99E-8/yr

And for Unit 2:

CLASS_3A_FREQUENCY = 0.027 ∗ (CDF-Class 2-Class 8–0.43*Category A)
= 0.027 * (5.81eE-05 – 5.03E-08 – 4.21E-07 – 0.43 * 4.46E-05) = 1.04E-6/yr
CLASS_3B_FREQUENCY = 0.0027 ∗ (CDF-Class 2-Class 8–0.43*Category A)
=0.0027 * ((5.81eE-05 – 5.03E-08 – 4.21E-07 – 0.43 * 4.46E-05) = 1.04E-7/yr

For this analysis, the associated containment leakage for Class 3A is $10L_a$ and for Class 3B is $35L_a$. These assignments are consistent with the guidance prescribed in EPRI/NEI Interim Guidance, Rev.4. [14]

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the NEI interim guidance [14], however, this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., overpressure). For this analysis, the frequency is determined from Release category K from the FNP Level 2 results. This equates to a frequency of 8.00E-06 for Unit 1 and 1.30E-05 for Unit 2.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. FNP Level 2 results assign release categories D and T to the containment bypass failure. The containment bypass failure frequency from the base model Level 2 results is the sum of release categories D and T. This results in 4.18E-07/year for Unit 1, and 4.21E-07/yr for Unit 2.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definition of Accident Classes defined in EPRI-TR-104285 and EPRI/NEI Interim Guidance, Rev.4. Table 5-2 summarizes these accident frequencies by Accident Class for FNP Unit 1 and Unit 2.

Table 5-2
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (FNP BASE CASE)

Accident Classes (Containment Release Type)	Description	FNP Unit 1 Frequency (per Rx-yr)	FNP Unit 2 Frequency (per Rx-yr)
1	No Containment Failure	2.96E-05	4.35E-05
2	Large Isolation Failures (Failure to Close)	2.39E-08	5.03E-08
3a	Small Isolation Failures (liner breach)	4.99E-07	1.04E-06
3b	Large Isolation Failures (liner breach)	4.99E-08	1.04E-07
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA
7	Failures Induced by Phenomena (Early and Late)	8.00E-06	1.30E-05
8	Bypass (Interfacing System LOCA)	4.18E-07	4.21E-07
CDF	**All CET End states (including very low and no release)**	**3.85E-05**	**5.81E-05**

5.2 STEP 2 - DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by NUREG/CR-4551 with adjustments made for the site demographic differences compared to the reference plant as described in Section 4.2, and summarized in Table 4.2-4. The results of applying these releases to the EPRI/NEI containment failure classification are shown below.

Class 1	=	148 person-rem (at 1.0L_a)	=	148 person-rem [(1)]
Class 2	=	2.26E+05 [(2)]		
Class 3a	=	148 person-rem x 10L_a	=	1.48E+03 person-rem [(3)]
Class 3b	=	148 person-rem x 35L_a	=	5.18E+03 person-rem [(3)]
Class 4	=	Not analyzed		
Class 5	=	Not analyzed		
Class 6	=	Not analyzed		
Class 7	=	1.73E+05 person-rem [(4)]		
Class 8	=	2.84E+05 person-rem [(5)]		

(1) The derivation is described in Section 4.2 for FNP. Class 1 is assigned the dose from the "No containment failure" APB from NUREG/CR-4551 (i.e., APB #6)

(2) Class 2 -Containment Isolation failures with a dose assigned from APB #2 (Early CF).

(3) The Class 3a and 3b releases are related to the Leakage rate as shown. This is consistent with the EPRI/NEI Interim Guidance, Rev.4.

(4) The Class 7 dose is assigned from APB #4 (Late CF).

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from APB #5 (Bypass).

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [2] containment failure classifications, and consistent with the NEI guidance [14] are provided in Table 5-3.

Table 5-3
FNP POPULATION DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	1.48E+02
2	Large Isolation Failures (Failure to Close)	2.26E+05
3a	Small Isolation Failures (liner breach)	1.48E+03
3b	Large Isolation Failures (liner breach)	5.18E+03
4	Small Isolation Failures (Failure to seal –Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g., dependent failures)	NA
7	Failures Induced by Phenomena (Early and Late)	1.73E+05
8	Bypass (Interfacing System LOCA)	2.84E+05

The above results when combined with the results presented in Table 5-2 yield the FNP baseline mean consequence measures for each accident class. These results are presented in Table 5-4a for Unit 1 and in table 5-4b for Unit 2.

Table 5-4a
FNP Unit 1
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS (I.E., REPRESENTATIVE OF ILRT DATA)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure [1]	2.96E-05	1.48E+02	4.37E-03
2	Large Isolation Failures (Failure to Close)	2.39E-08	2.26E+05	5.40E-03
3a	Small Isolation Failures (liner breach)	4.99E-07	1.48E+03	7.39E-04
3b	Large Isolation Failures (liner breach)	4.99E-08	5.18E+03	2.59E-04
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	8.00E-06	1.73E+05	1.38E+00
8	Bypass (Interfacing System LOCA)	4.18E-07	2.84E+05	1.19E-01
CDF	**All CET End states (including very low and no release)**	3.85E-05		1.513

[1] Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Category 3a and 3b include failures of containment to meet the Technical Specification leak rate.

Table 5-4b
FNP Unit 2
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS (I.E., REPRESENTATIVE OF ILRT DATA)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure [(1)]	4.35E-05	1.48E+02	6.43E-03
2	Large Isolation Failures (Failure to Close)	5.03E-08	2.26E+05	1.14E-02
3a	Small Isolation Failures (liner breach)	1.04E-06	1.48E+03	1.54E-03
3b	Large Isolation Failures (liner breach)	1.04E-07	5.18E+03	5.38E-04
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	1.30E-05	1.73E+05	2.25E+00
8	Bypass (Interfacing System LOCA)	4.21E-07	2.84E+05	1.20E-01
CDF	**All CET End states (including very low and no release)**	5.81E-05		2.388

(1) Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Category 3a and 3b include failures of containment to meet the Technical Specification leak rate.

The FNP dose compares favorably with other locations given the relative population densities surrounding each location:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	[6]
Peach Bottom	6.2	[16]
Farley Unit 2	2.4	[Table 5.4b]
Farley Unit 1	1.5	[Table 5.4a]
Crystal River	1.4	[17]

Based on the risk values from Tables 5-4a and 5-4b, the percent risk contribution ($\%Risk_{BASE}$) for Class 3 is as follows:

$$\%Risk_{BASE} = (CLASS3a_{BASE} + CLASS3b_{BASE}) / Total_{BASE} \times 100$$

Where, for Unit 1:

$CLASS3a_{BASE}$ = 7.39E-4 person-rem/year [Table 5-4a]

$CLASS3b_{BASE}$ = 2.59E-4 person-rem/year [Table 5-4a]

$TOTAL_{BASE}$ = 1.513 person-rem/yr [Table 5-4a]

$\%Risk_{BASE}$ = [(7.39E-4 + 2.59E-4) / 1.513] *100 = (9.98E-4) / 1.513

$\%Risk_{BASE}$ = 0.07%

And for Unit 2:

$CLASS3a_{BASE}$ = 1.54E-3 person-rem/year [Table 5-4b]

$CLASS3b_{BASE}$ = 5.38E-4 person-rem/year [Table 5-4b]

$TOTAL_{BASE}$ = 2.388 person-rem/yr [Table 5-4b]

$\%Risk_{BASE}$ = [(1.54E-3 + 5.38E-4) / 2.388] *100 = (2.08E-3) / 2.388

$\%Risk_{BASE}$ = 0.09%

Therefore, the Total Type A 3/10-years ILRT interval risk contribution of leakage, represented by Class 3a and Class 3b accident scenarios is 0.07% for Unit 1 and 0.09% for Unit 2 in the base case analysis.

5.3 STEP 3 - EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEARS

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to a fifteen-year interval. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case is assumed to apply to a 3-year interval (i.e., a simplified representation of a 3-in-10 interval).

Risk Impact due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval, (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3 sequences is impacted. Therefore, for Class 3 sequences, the risk contribution is changed based on the NEI guidance as described in Section 4.4 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 5-5a for Unit 1 and Table 5-5b for Unit 2.

Table 5-5a
FNP Unit 1
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure	2.83E-05	1.48E+02	4.18E-03
2	Large Isolation Failures (Failure to Close)	2.39E-08	2.26E+05	5.40E-03
3a	Small Isolation Failures (liner breach)	1.67E-06	1.48E+03	2.47E-03
3b	Large Isolation Failures (liner breach)	1.67E-07	5.18E+03	8.63E-04
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	8.00E-06	1.73E+05	1.38E+00
8	Bypass (Interfacing System LOCA)	4.18E-07	2.84E+05	1.19E-01
CDF	**All CET End states (including very low and no release)**	3.85E-05		1.516

Table 5-5b
FNP Unit 2
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure	4.08E-05	1.48E+02	6.04E-03
2	Large Isolation Failures (Failure to Close)	5.03E-08	2.26E+05	1.14E-02
3a	Small Isolation Failures (liner breach)	3.45E-06	1.48E+03	5.11E-03
3b	Large Isolation Failures (liner breach)	3.45E-07	5.18E+03	1.79E-03
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	1.30E-05	1.73E+05	2.25E+00
8	Bypass (Interfacing System LOCA)	4.21E-07	2.84E+05	1.20E-01
CDF	**All CET End states (including very low and no release)**	5.81E-05		2.393

Based on the risk values from Tables 5-5a and 5-5b, the percent risk contribution ($\%Risk_{10}$) for Class 3 is as follows:

$$\%Risk_{10} = (CLASS3a_{10} + CLASS3b_{10}) / Total_{10} \times 100$$

Where, for Unit 1:

$CLASS3a_{10}$ = 2.47E-3 person-rem/year [Table 5-5a]

$CLASS3b_{10}$ = 8.63E-4 person-rem/year [Table 5-5a]

$TOTAL_{10}$ = 1.516 person-rem/yr [Table 5-5a]

$\%Risk_{10}$ = [(2.47E-3 + 8.63E-4) / 1.516] *100 = (3.33E-3) / 1.516

$\%Risk_{10}$ = 0.22%

And for Unit 2:

$CLASS3a_{10}$	= 5.11E-3 person-rem/year [Table 5-5b]
$CLASS3b_{10}$	= 1.79E-3 person-rem/year [Table 5-5b]
$TOTAL_{10}$	= 2.393 person-rem/yr [Table 5-5b]
$\%Risk_{10}$	= [(5.11E-3 + 1.79E-3) / 2.393] *100 = (6.90E-3) / 2.393
$\%Risk_{10}$	= 0.29%

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3a and Class 3b accident scenarios is 0.22% for Unit 1 and 0.29% for Unit 2.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value as described in section 4.4. The results for this calculation are presented in Table 5-6a for Unit 1 and in Table 5-6b for Unit 2.

Table 5-6a
FNP Unit 1
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure (1)	2.73E-05	1.48E+02	4.05E-03
2	Large Isolation Failures (Failure to Close)	2.39E-08	2.26E+05	5.40E-03
3a	Small Isolation Failures (liner breach)	2.50E-06	1.48E+03	3.70E-03
3b	Large Isolation Failures (liner breach)	2.50E-07	5.18E+03	1.30E-03
4	Small Isolation Failures (Failure to seal --Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	8.00E-06	1.73E+05	1.38E+00
8	Bypass (Interfacing System LOCA)	4.18E-07	2.84E+05	1.19E-01
CDF	**All CET End states (including very low and no release)**	3.85E-05		1.517

Table 5-6b
FNP Unit 2
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure (1)	3.89E-05	1.48E+02	5.76E-03
2	Large Isolation Failures (Failure to Close)	5.03E-08	2.26E+05	1.14E-02
3a	Small Isolation Failures (liner breach)	5.19E-06	1.48E+03	7.68E-03
3b	Large Isolation Failures (liner breach)	5.19E-07	5.18E+03	2.69E-04
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7	Failures Induced by Phenomena (Early and Late)	1.30E-05	1.73E+05	2.25E+00
8	Bypass (Interfacing System LOCA)	4.21E-07	2.84E+05	1.20E-01
CDF	**All CET End states (including very low and no release)**	5.81E-05		2.396

Based on the values from Tables 5-6a and 5-6b, the Type A 15-year test frequency percent risk contribution (%Risk $_{15}$) for Class 3 is as follows:

$$\%Risk_{15} = (CLASS3a_{15} + CLASS3b_{15}) / Total_{15} \times 100$$

Where, for Unit 1:

$CLASS3a_{15}$ = 3.70E-3 person-rem/year [Table 5-6a]

$CLASS3b_{15}$ = 1.30E-3 person-rem/year [Table 5-6a]

$TOTAL_{15}$ = 1.517 person-rem/yr [Table 5-6a]

$\%Risk_{15}$ = [(3.70E-3 + 1.30E-3) / 1.518] *100 = (5.00E-3) / 1.517

$\%Risk_{15}$ = 0.33%

And for Unit 2:

$CLASS3a_{15}$ = 7.68E-3 person-rem/year [Table 5-6b]

$CLASS3b_{15}$ = 2.69E-3 person-rem/year [Table 5-6b]

$TOTAL_{15}$ = 2.396 person-rem/yr [Table 5-6b]

$\%Risk_{15}$ = [(7.68E-3 + 2.69E-3) / 2.396] *100 = (1.04E-2) / 2.396

$\%Risk_{B15}$ = 0.43%

Therefore, the Total Type A 15-year ILRT interval risk contribution of leakage, represented by Class 3a and Class 3b accident scenarios is 0.33% for Unit 1 and 0.43% for unit 2.

In summary, the results above show that the percent contribution from risk due to ILRT-averted leakage scenarios is small (i.e., less than 0.5%) in all cases. It is also appropriate to provide a comparison of the change in the total integrated plant risk. The percent increase on the total integrated plant risk when the ILRT is extended from 10 years to 15 years is computed as follows:

$$\%TOTAL_{10-15} = [(TOTAL_{15} - TOTAL_{10}) / TOTAL_{10}] \times 100$$

Where, for Unit 1:

$TOTAL_{10}$ = 1.516 person-rem/year [Table 5-5a]

$TOTAL_{15}$ = 1.517 person-rem/year [Table 5-6a]

$\%TOTAL_{10-15} = [(1.516 - 1.517) / 1.516] \times 100$

$\%TOTAL_{10-15} = 0.07\%$

And for Unit 2:

$TOTAL_{10}$ = 2.393 person-rem/year [Table 5-5b]

$TOTAL_{15}$ = 2.396 person-rem/year [Table 5-6b]

$\%TOTAL_{10-15} = [(2.396 - 2.393) / 2.393] \times 100$

$\%TOTAL_{10-15} = 0.13\%$

Therefore, the risk impact on the total integrated plant risk for an ILRT extension from 10 to 15 years is only 0.07% for Unit 1 and 0.13% for Unit 2.

5.4 STEP 4 - DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY (LERF)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the NEI guidance, 100% of the Class 3b contribution would be considered LERF. For Farley, however, the Class 3b radionuclide release person-rem is significantly less than a typical LERF contributor as can be seen by comparing the relative population dose for Class 3b to that of Class 2 (5.18E+03 person-rem / 2.26E+05 person-rem) or 2.3%. Additionally, the Farley calculated dose for Accident Class 3b is also much lower than that calculated in previous submittals for an ILRT extension (e.g. 4.94E7 person-rem for IP3 [6] and 3.45E4 person-rem for Crystal River [17] - both were NRC approved).

Reg. Guide 1.174 [3] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

For Unit 1, 100% of the frequency of Class 3B sequences (consistent with the NEI Guidance methodology) can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension. Based on a ten-year test interval from Table 5-5a, the Class 3b frequency is 1.67E-07/yr; and, based on a fifteen-year test interval from Table 5-6a, it is 2.50E-07. Thus, increasing the ILRT test interval from 10 to 15 years results in an additional 8.36E-8/yr increase in the overall probability of LERF due to Class 3b sequences. As can be seen, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF for FNP Unit 1 is below the threshold criteria for a very small change.

For Unit 2, based on a ten-year test interval from Table 5-5b, the Class 3b frequency is 3.45E-07/yr; and, based on a fifteen-year test interval from Table 5-6b, it is 5.19E-07. Thus, increasing the ILRT test interval from 10 to 15 years results in an additional 1.73E-7/yr increase in the overall probability of Class 3b sequences. Realistically though, only a fraction of the Class 3b frequency would actually have the potential to be considered as large enough for LERF at Farley. Figure 5-1 shows a representative population dose frequency distribution for Class 3b with the calculated mean value of 5180 person-rem (based on a characteristic 35La release magnitude) using a lognormal distribution and with an assumed range factor of 10. As can be seen, only a small probability would be associated with a release magnitude that would be categorized as sufficiently Large for a LERF release. The reverse cumulative distribution plot shown in Figure 5-2 bears out the fact that the probability of a large release (i.e., assumed to be represented by a magnitude approaching 5E4 person-rem or higher) is quite small in this example. Consequently, even with a larger assumed error factor, it can be conservatively stated that no more than 10% of the Class 3b releases could have the potential to be LERF for Farley. Therefore, a factor of 10 reduction can be made to the calculated EPRI Class 3b scenarios in reporting a more realistic assessment of the LERF contribution from the ILRT extension for FNP Unit 2. This correlates to an estimated LERF increase of 1.73E-8/yr assuming as described above, that 10% of the frequency of Class 3B sequences can be used as a first-order estimate to approximate a more realistic potential increase in LERF from the ILRT interval extension. As can be seen, the estimated change in LERF for FNP Unit 2 is then also below the threshold criteria for a very small change.

Additionally, the frequency of the Class 3b for Unit 2 is relatively higher that the Unit 1 frequency. This is chiefly due to a design difference between Unit 1 and Unit 2 in the Service Water System. The Service Water Pumps for Unit 2 are modeled in the FNP PRA as requiring a booster pump in their Lube and Cooling System to provide a filtered source of cooling water. These booster pumps are the safety-related source of Lube and Cooling for the Service Water Pumps. The non-safety-related part of the Lube and Cooling system is provided through a cyclone separator and is not powered from the emergency power source. Since each Train has only one booster pump, the loss of a booster pump has significant affects in the PRA model. The PRA model is very conservative in that it does not credit any operator actions that would restore the lube and cooling if the booster pumps failed. The Operator actions that can be taken in a timely manor to insure SW pumps continue to operate include: 1) on a loss of AC event, supplying power from Unit 1 to the Unit 2 non safety related portion of the Lube and Cooling System that does not require the booster pumps. 2) Monitor the Service Water Pump seal leak-off to insure adequate flow past the shaft packing. The design of the Service Water pump is such that if the Lube and Cooling water supply is lost from the non safety related supply and from the booster pumps, the process flow of Service Water through the pump would provide flow through the Lube and Cooling channels and out past the pump packing rings. These are actions that can be performed by the System Operator in a reasonable time and would insure that the Service Water pumps continue to operate until the normal Lube and Cooling water supply can be restored. The conservative modeling in the PRA model leads to the higher event frequency. FNP is in the process of performing design changes on the Unit 2 Service Water pumps that will remove the requirement for the booster pumps and thus eliminate a failure mechanism from the PRA model. FNP has completed this design change on one of the five Unit 2 Service Water Pumps and has scheduled installation of that pump in March 2002. The remaining Service Water Pumps will be modified and installed over the next five years. After the design changes are completed and the PRA model is adjusted, the frequency of the Class 3b events for Unit 2 will be substantially reduced making the frequency very similar to Unit 1.

Figure 5-1
POPULATION DOSE (PERSON-REM) FREQUENCY DISTRIBUTION FOR ACCIDENT CLASS 3B

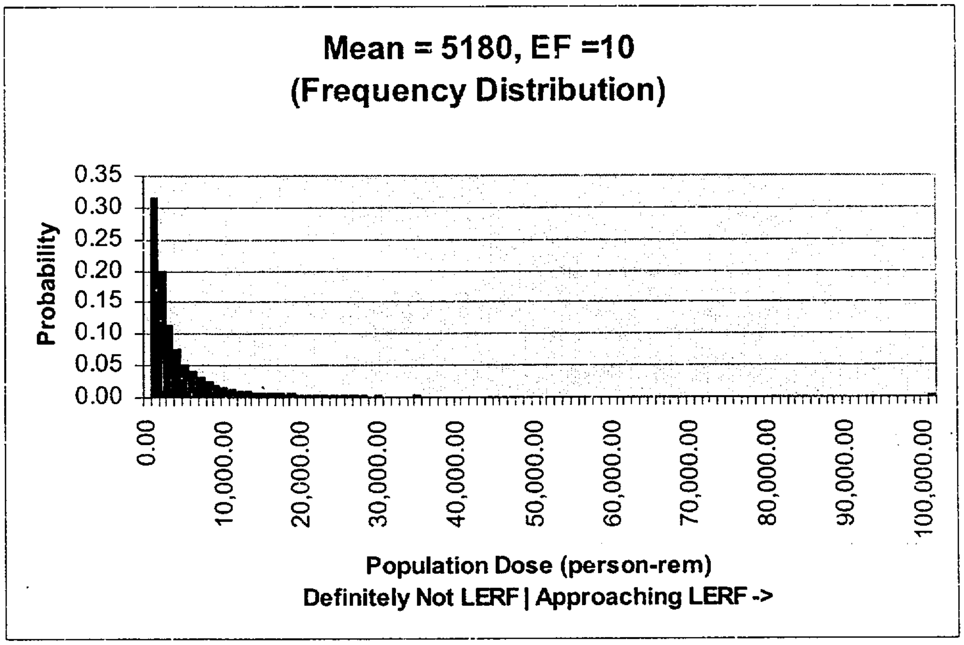
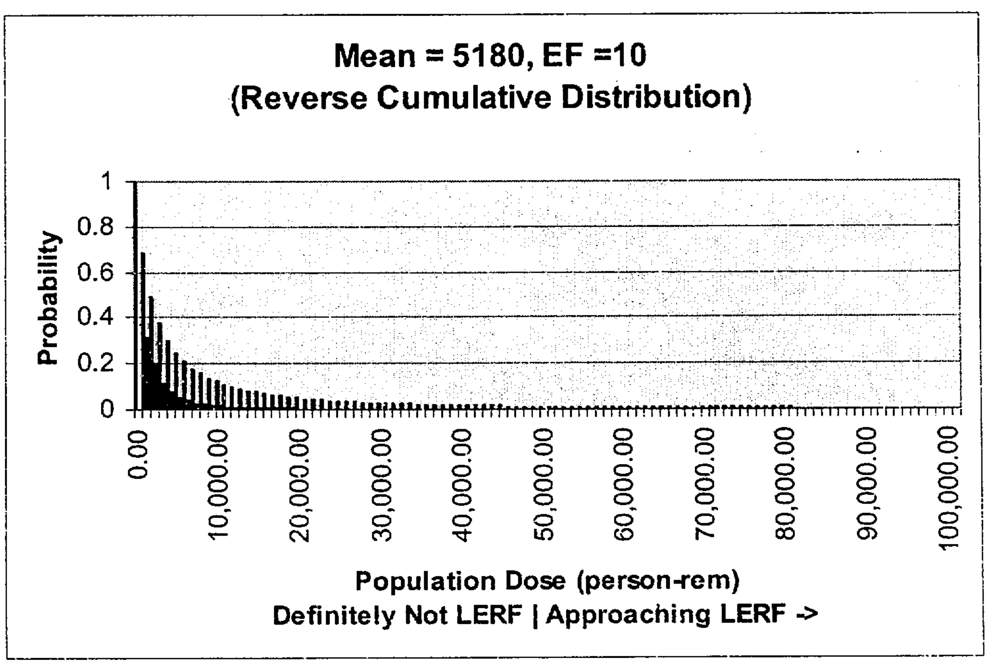


Figure 5-2
POPULATION DOSE (PERSON-REM) REVERSE CUMULATIVE DISTRIBUTION

FOR ACCIDENT CLASS 3B



5.5 IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY (CCFP)

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in NEI Interim Guidance [14]. The NRC has previously accepted similar calculations (Ref: Indian Point 3 License Amendment) as the basis for showing that the proposed change is consistent with the defense in depth philosophy.

$$CCFP = [1 - (\text{Class 1 frequency} + \text{Class 3a frequency}) / CDF] * 100\%$$

$$\Delta CCFP = CCFP_{15} - CCFP_{10} = 0.22\% \text{ for Unit 1, and } 0.30\% \text{ for Unit 2}$$

This change in CCFP of less than 0.5% for both units is judged to be insignificant.

5.6 RESULTS SUMMARY

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

1. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem) represented by Class 3 accident scenarios is 0.22% for unit 1 and 0.29% for Unit 2. When the ILRT interval is 15 years, the risk contribution of leakage represented by Class 3 accident scenarios is 0.33% for unit 1 and 0.43% for Unit 2. The change in the percent contribution to risk of just 0.11% or 0.14% is judged to be insignificant.

2. The total integrated risk is 1.516 person-rem/yr for Unit 1 and 2.393 person-rem/yr for Unit 2 when the ILRT interval is ten years. The total integrated risk is 1.517 person-rem/yr for Unit 1 and 2.396 person-rem/yr for Unit 2 when the ILRT interval is fifteen years. Consequently, the total integrated increase in risk contribution from extending the ILRT test frequency from the current one-per-10-year interval to once-per-15 years is less than 0.10% for both of the FNP units. This is also judged to be insignificant.

3. For Unit 1: Exactly consistent with the NEI Guidance methodology, the Class 3b frequency can be conservatively characterized as LERF (even though the release magnitude is about two orders of magnitude less than other LERF categories). The risk increase in Class 3b frequency from extending the ILRT test frequency from the current once-per-10 year interval to once-per-15 years is 8.36E-8/yr. Even with all of this frequency very conservatively considered as LERF, this is determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

4. For Unit 2: The change in the Class 3b frequency is 1.73E-7/yr. However, based on the analysis presented in Section 5.4, it can assumed for Farley that 10% of the frequency of Class 3B sequences represents a less conservative first-order

estimate to approximate the potential increase in LERF from the ILRT interval extension. Consequently, the risk increase in LERF from extending the ILRT test frequency from the current once-per-10 year interval to once-per-15 years correlates to 1.73E-8/yr. This is also determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

5. The change in the conditional containment failure frequency from the current once-per-10 year interval to once-per-15 years is 0.22% and 0.30%, respectively, for FNP Unit 1 and FNP Unit 2. Though no official acceptance criteria exist for this risk metric, it is also judged to be very small.

These results are summarized in Table 5-7a and 5-7b

Table 5-7a
FNP Unit 1
SUMMARY OF RISK IMPACT ON TYPE A ILRT TEST FREQUENCY

Category	Risk Impact (Base) [(2)]	Risk Impact (10-years) [(3)]	Risk Impact (15-years) [(4)]
Class 3a and 3b [(1)]	0.07% of integrated value 9.98E-4 person-rem/yr	0.22% of integrated value 3.33E-3 person-rem/yr	0.33% of integrated value 5.00E-3 person-rem/yr
Total Integrated Risk	1.513 person-rem/year	1.516 person-rem/year	1.517 person-rem/year
Class 3b	4.99E-8/yr	1.67E-7/yr	2.50E-7/yr
CCFP	22.03%	22.34%	22.55%

(1) Only accident sequences increased by a change in Type A test frequency are evaluated. These are sequences 3a and 3b.

(2) FNP IPE baseline values based on a 3-in-10 (simplified to 1-in-3) year interval.

(3) Type A ILRT test interval of 1-in10 years.

(4) Type A ILRT test interval of 1-in15 years.

Table 5-7b
FNP Unit 2
SUMMARY OF RISK IMPACT ON TYPE A ILRT TEST FREQUENCY

Category	Risk Impact (Base) [(2)]	Risk Impact (10-years) [(3)]	Risk Impact (15-years) [(4)]
Class 3a and 3b [(1)]	0.09% of integrated value 2.07E-3 person-rem/yr	0.29% of integrated value 6.90E-3 person-rem/yr	0.43% of integrated value 1.04E-2 person-rem/yr
Total Integrated Risk	2.388 person-rem/year	2.393 person-rem/year	2.396 person-rem/year
Class 3b	1.04E-7/yr	3.45E-7/yr	5.19E-7/yr
CCFP	23.38%	23.79%	24.09%

(1) Only accident sequences increased by a change in Type A test frequency are evaluated. These are sequences 3a and 3b.

(2) FNP IPE baseline values based on a 3-in-10 (simplified to 1-in-3) year interval.

(3) Type A ILRT test interval of 1-in10 years.

(4) Type A ILRT test interval of 1-in15 years.

Section 6
CONCLUSIONS

Based on the results from Section 5, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test frequency from ten-years to fifteen years:

- Reg. Guide 1.174 [3] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once-per-ten years to once-per-fifteen years is very conservatively estimated as 8.36E-8/yr for Unit 1 (using the NEI guidance as written) and more realistically estimated at 1.73E-8/yr for Unit 2 (with a slight variation to the NEI guidance). As such, the estimated changes in LERF are determined to be "very small" using the acceptance guidelines of Reg. Guide 1.174.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years, measured from the percent contribution to risk or from the increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is less than 0.2% in all cases. Therefore, the risk impact when compared to other severe accident risks is negligible.

- The change in the conditional containment failure frequency from the current once-per-10 year interval to once-per-15 years is 0.22% and 0.30%, respectively, for FNP Unit 1 and FNP Unit 2. Though no official acceptance criteria exist for this risk metric, it is also judged to be very small.

Therefore, increasing the ILRT interval from 10 to 15 years is considered to be insignificant since it represents a very small change to the Farley Nuclear Plant risk profile.

Previous Assessments

The NRC in NUREG-1493 [4] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment.

The findings for Farley Nuclear Plant confirm these general findings on a plant specific basis considering the severe accidents evaluated for FNP, the FNP containment failure modes, and the local population surrounding Farley Nuclear Plant.

Section 7

REFERENCES

[1] *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,* NEI 94-01, July 1995.

[2] *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,* EPRI, Palo Alto, CA EPRI TR-104285, August 1994.

[3] *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Regulatory Guide 1.174, July 1998.

[4] *Performance-Based Containment Leak-Test Program,* NUREG-1493, September 1995.

[5] *Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check',* NUREG-1273, April 1988.

[6] Letter from R.J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 18, 2001.

[7] United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

[8] *Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™*, EPRI, Palo Alto, CA TR-105189, Final Report, May 1995.

[9] *Evaluation of Severe Accident Risks: Surry Unit 1,* Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, October 1990.

[10] *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.

[11] *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.

[12] *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.

[13] *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG -1150, December 1990.

[14] *NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, November 2001.

[15] Anthony R. Pietrangelo, *One-time extensions of containment integrated leak rate test interval – additional information*, NEI letter to Administrative Points of Contact, November 30, 2001.

[16] Letter from J.A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DPR-56, LAR-01-00430, dated May 30, 2001.

[17] Letter from D.E. Young (Florida Power, Crystal River) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.

[18] United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.

[19] E-Mail from D.E.McCoy (Southern Co.) to J.R Gabor (ERIN) *Plant Specific Inputs requested by ERIN for FNP Frequency Extension,* January 8, 2002.