NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

April 22, 2002

10 CFR Part 50,
Section 50.90

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

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request for
Risk-Informed Technical Specification Change
Regarding Five Year Extension of Type A Test Interval

Pursuant to and in accordance with the requirements of 10 CFR Part 50, Sections 50.59 and 50.90 Nuclear Management Company, LLC (NMC) hereby requests a change to the Technical Specifications (TS), Appendix A of Operating License DPR-22, for the Monticello Nuclear Generating Plant.

The proposed change will revise Monticello TS to permit a one-time five-year extension, to no later than March 2008, of the ten-year performance based Type A test interval established in NEI 94-01, "Nuclear Energy Institute Industry Guideline for Implementing Performance–Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995.

This TS change has been prepared in accordance with the guidance provided in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis."

A plant-specific, risk-based evaluation has been performed in support of this one-time exception to extend the Type A test interval. This evaluation uses the latest Monticello probabilistic safety assessment (PSA) models to estimate the changes in risk associated with increasing the Type A testing interval. This risk assessment is consistent with current PSA best practices. The release category and person-rem information is based on design basis leakage evaluations and extrapolation of the release category information using a modeling approach that is described in Exhibit B.

This license amendment application represents a cost-beneficial licensing action. The Type A test imposes significant expense on NMC while the safety benefit of performing the Type A test within 10 years, versus 15 years, is minimal. This request is similar to license amendments authorized by the NRC on August 30, 2001 (ADAMS Accession Number ML012190219) for the Crystal River Nuclear Plant, Unit 3, and on February 20, 2002 (ADAMS Accession Number ML0205603210) for the Edwin I. Hatch Nuclear Plant, Unit 1.

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Exhibit A contains the Proposed Changes, Reasons for Change, a Safety Implication, a Determination of No Significant Hazards Consideration and an Environmental Assessment. Exhibit B contains the risk assessment for Monticello regarding ILRT (Type A) extension. Exhibit C contains current Monticello Technical Specification pages marked up with the proposed changes. Exhibit D contains revised Monticello Technical Specification pages.

This application has been reviewed by the Monticello Operations Committee and the Offsite Review Committee. A copy of this submittal, along with the Determination of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR 50.91(b)(1). In addition, a separate License Amendment Request, which affects the same TS pages as this request, is being submitted for NRC review and approval. NMC request that the NRC notify Monticello prior to issuance of the License Amendment to ensure that the most current Monticello TS pages are being issued.

Nuclear Management Company, LLC requests NRC approval of this Technical Specification change by December 1, 2002, to facilitate planning and scheduling for the next refueling outage, which is currently scheduled to begin on April 26, 2003. NMC request a period of up to 60 days following receipt of the license amendment to implement the changes.

If you have any questions regarding this License Amendment Request please contact Doug Neve, Licensing Manager, at (763) 295-1353.

Jeffrey S. Forbes
Site Vice President
Monticello Nuclear Generating Plant

Subscribed to and sworn before me this 22 day of april, 2002.

Kathryn I Kleine
Notary

KATHRYN I. KLEINE
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

Attachments:	Exhibit A –	Evaluation of Proposed Changes to the Monticello Technical Specifications
	Exhibit B –	PSA Assessment
	Exhibit C –	Current Monticello Technical Specification Pages Marked Up With Proposed Changes
	Exhibit D -	Revised Monticello Technical Specification Pages

cc: Regional Administrator-III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
Minnesota Department of Commerce
J. Silberg, Esq.

EXHIBIT A

Evaluation of Proposed Changes to the Monticello Technical Specifications

Introduction

Pursuant to and in accordance with the Code of Federal Regulations, Title 10, Parts 50.59 and 50.90, Nuclear Management Company, LLC (NMC) hereby requests changes to Appendix A, of Facility Operating License DPR-22, Technical Specifications (TS) for the Monticello Nuclear Generating Plant. The proposed license amendment requests a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests, as specified by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and endorsed by 10 CFR 50, Appendix J, Option B. The one-time exception applies to the requirement of NEI 94-01 to perform an integrated leak rate test (ILRT) at a frequency of up to 10 years, with an allowance for a 15 month extension.

Background

The Monticello Nuclear Generating Plant current 10-year Type A test interval ends in March 2003. In order to meet the interval requirements of NEI 94-01, this test must be performed during Refueling Outage 21, which is currently scheduled to commence on April 26, 2003. By granting the proposed one-time exception, Monticello would benefit by not having to perform the Type A test for an additional five years. Cost savings are estimated at $300,000 for elimination of the actual performance of the test. In addition, up to forty-eight hours of critical path outage time can be eliminated by not performing the Type A test. The critical path outage time is estimated at a savings of $500,000.

The NMC is aware of an ongoing industry/NRC initiative to modify the existing performance-based leakage testing guidance to extend the maximum Type A test interval. Therefore, the requested exception is limited to only five years for Monticello, which is considered an adequate amount of time to complete the testing guidance initiative change.

Proposed Change and Reason for Change

Monticello TS Surveillance Requirement (SR) 4.7.A.2.b, "Primary Containment Integrity" currently states:

> Perform required visual examinations and leakage rate testing for Type A containment integrated leakage rate tests in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and Regulatory Guide 1.163 dated September 1995. Perform Type B and C tests in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.

EXHIBIT A

Evaluation of Proposed Changes to the Monticello Technical Specifications

Monticello TS SR 4.7.A.2.b is being revised to add the following phrase after the end of the first sentence:

- "as modified by the following exception:"

And include the following exception to NEI 94-01:

- NEI 94-01 – 1995, Section 9.2.3: The first Type A test performed after the March 1993 Type A test shall be performed no later than March 2008.

This proposed amendment to the Monticello TS takes a one-time exception to the 10-year frequency of the performance-based leakage rate testing SR for Type A test as required by NEI 94-01. The exception is to allow ILRT testing within fifteen years from the last ILRT, which was performed in March 1993.

This application represents a cost beneficial licensing change. The ILRT imposes significant expense to the plant while the differential safety benefit of performing it within ten years, versus fifteen years, is minimal.

Revised Technical Specification Bases pages are also included in this submittal.

Safety Implication of the Proposed Change

Implementing 10 CFR 50, Appendix J, Option B:

Primary containment provides an essentially leak-tight barrier against the uncontrolled release of radioactivity into the environment following a design basis accident. The testing requirements of 10 CFR Part 50, Appendix J, provide assurance that leakage from the primary containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the Technical Specifications. The limitation of containment leakage provides assurance that the containment will perform its design function following a design basis accident.

Effective October 26, 1995, 10 CFR Part 50, Appendix J, was revised to allow licensees to choose to perform containment leakage testing under Option A, "Prescriptive Requirements" or Option B, "Performance-Based Requirements." On April 3, 1996, License Amendment 95 for Monticello was issued to permit implementation of 10 CFR Part 50, Appendix J, Option B, for the Type A containment integrated leakage rate test as modified by approved exemptions, and Regulatory Guide (RG) 1.163. RG 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8, subject to several regulatory positions in the guide.

Exceptions to the requirements of RG 1.163, are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states:

> The Regulatory Guide or other implementing document used by a licensee, or other applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant Technical Specifications. The submittal for Technical Specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

Therefore, this application does not require an exemption to 10 CFR 50, Appendix J, Option B.

The adoption of the Option B performance-based containment leakage rate testing for Type A testing 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 changes to Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

The allowed frequency for Type A testing was based upon a generic evaluation documented in NUREG-1493. NUREG-1493 made the following observations with regard to decreasing the test frequency:

- "Reducing the Type A (ILRT) testing frequency to one per twenty years was found to lead to an 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."

The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than $1.0L_a$ and consideration of the performance factors in NEI 94-01, Section 11.3). Based on the results of the March 1993, and November 1989, ILRTs, the current interval for Monticello is once every ten years.

Regulatory Guide 1.163 Containment Visual Examinations

NMC has established procedures for performing visual examination of the accessible surfaces of the containment for detection of structural problems. RG 1.163, Regulatory Position C.3 specifies that these examinations should be conducted prior to initiating a Type A test and during two other outages before the next Type A test if the interval for the Type A test has been extended to ten years, in order to allow for early detection of evidence of structural deterioration. These visual examinations are being done, with no significant defects noted to date.

IWE and IWL Containment Inspection Program Activities

Pursuant to 10 CFR 50.55a(g)(6)(ii)(B), NMC has developed a Containment Inspection Program for the Monticello Nuclear Generating Plant. The Containment Inspection Program was established in 1995, in accordance with Subsections IWE and IWL of ASME Section XI, 1992 Edition, to assure detection of deterioration affecting containment integrity. The Monticello containment is a free-standing steel containment, to which only the requirements of Subsection IWE apply.

The Monticello IWE Program meets the requirements of the 1992 Edition with the 1992 Addenda of ASME Section XI. The First Ten-Year Containment Inspection Interval started September 9, 1996 with the first period examinations completed by September 9, 2001 (as required by 10 CFR 50.55a(g)(6)(ii)(B)(1)). The three inspection periods during the containment inspection interval are as follows:

First Period:	September 9, 1996 - September 8, 2001
Second Period:	September 9, 2001 - September 8, 2005
Third Period:	September 9, 2005 - September 8, 2008

The ASME IWE inspections include the interior liner and the exterior concrete surfaces. In general, the areas and items subject to inspection include the accessible class MC pressure retaining containment surface areas, including structural attachments and penetrations, seals, gaskets, moisture barriers, pressure retaining bolting, and Class MC supports. Exceptions taken to the ASME Section XI requirements have been documented and approved by the NRC as requests for relief. Inaccessible areas are evaluated for degradation when conditions in accessible areas indicate the presence of or result in degradation not meeting the established acceptance standards.

EXHIBIT A

Evaluation of Proposed Changes to the Monticello Technical Specifications

The most recent visual inspection of the Monticello containment was performed in 2001. This visual inspection was performed by qualified individuals and resulted in no adverse conditions being identified.

The Monticello containment design includes a steel drywell and suppression chamber with interconnecting vent pipes with bellows. The bellows assembly connects the suppression chamber to the vent lines that allows for differential movement between the drywell and the suppression chamber. The controlled atmosphere of the suppression chamber (i.e., nitrogen atmosphere which is maintained during power operation), the protective cover over the bellows and the location ensure an environment that is resistant to stress corrosion cracking.

To assure comprehensive inspection of the containment, the Containment Inspection Program has been integrated with visual inspection activities performed in conjunction with Maintenance Rule activities, as well as with Type A testing. The integration of these inspection activities provides a consistent and effective approach for assessing the condition of the containment and assuring detection of degradation. There will be no change to the schedule for the Containment Inspection Program activities as a result of this license amendment application.

Plant Operational Performance

Monticello is a boiling water reactor contained in a Mark I containment. During power operation, the primary containment atmosphere is inerted with nitrogen to ensure that no external sources of oxygen are introduced into containment. The containment inerting system is used during the initial purging of the primary containment prior to power operation and provides a supply of makeup nitrogen to maintain primary containment oxygen concentration within Technical Specification limits. As a result, the primary containment is maintained at a slightly positive pressure during power operation. During power operation, instrument air system (i.e., nitrogen) leaks occur from pneumatically-operated valves inside the containment which gradually pressurize the primary containment. Primary containment pressure is monitored in the control room. The primary containment atmosphere is periodically vented in order to maintain containment pressure within an acceptable operating range. This cycling of the primary containment pressure during operation amounts to a periodic integrated pressure test of the containment at a low differential pressure. Although this cycling does not challenge the structural and leak tight integrity of the primary containment system at post-accident pressure, it provides assurance that a gross containment leakage that may develop during power operation will be detected. This feature is a complement to visual inspection of the interior and exterior of the containment structure for those areas that may be inaccessible for visual examination. In the event pressurization does not occur, a leakage path may be present. Plant operators are aware of the implications of lack of pressurization during power operation. Following approval of this license amendment application, administrative controls will be established to monitor containment depressurization activities and evaluate trends (e.g., frequency, duration) for indication of changes to containment leakage.

Plant Specific Risk Assessment for the Extended ILRT Test Interval

A plant-specific risk assessment was performed in support of the one-time exception to extend the Type A test, for Monticello, from once in ten years to once in fifteen years. This risk assessment was performed in accordance with the guidelines set forth in NEI 94-01. A copy of this plant-specific risk assessment is provided in Exhibit B.

The plant-specific risk assessment uses the latest Monticello Level 1 and Level 2 probabilistic safety assessment (PSA) models to estimate the changes in risk associated with increasing the Type A testing interval. The release category and person-rem information is based on design basis leakage evaluations and extrapolation of the release category information using a modeling approach that is described in the Risk Assessment in Exhibit B. This assessment uses the methodology described in Electric Power Research Institute (EPRI) Topical Report (TR)-104285 to estimate plant risk on specific accident sequences impacted by Type A testing.

The guidance in NRC Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," dated July 1998, on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, was used in evaluating the results of this risk assessment.

The plant-specific risk assessment determined that a change in Type A test frequency from ten years to fifteen years will have an extremely small change in population dose consequences. Specifically, the proposed Type A test frequency change from ten to fifteen years will result in a 0.6 percent increase in total integrated plant risk.

Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6 per year and increases in Large Early Release Frequency (LERF) below 1E-7 per year. The proposed extension of the Type A test interval does not have an impact on CDF. Therefore, the change in LERF provides the appropriate assessment of the change in risk associated with the proposed change. The increase in LERF resulting from the proposed Type A test frequency change from ten to fifteen years is 7.1E-8/yr. Therefore, based on this risk assessment, the proposed change to the Type A test frequency does not represent a risk significant change.

Regulatory Guide 1.174 also encourages the use of risk analysis techniques to ensure and demonstrate that a proposed change is consistent with the defense-in-depth philosophy. This philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. For the proposed Type A test frequency extension from ten to fifteen years, the change in conditional containment failure probability was determined to be 0.4 percent. Thus, these changes are small and the defense-in-depth is maintained.

10 CFR 50 Appendix J, Option B Test Information:

A Type A test can detect containment leakage due to a loss of structural capability. All other sources of containment leakage detected in Type A test analyses can be detected by the Type B and C tests.

Previous Type A tests confirmed that the Monticello reactor containment structure has extremely low leakage and represents an insignificant potential risk contributor to increased containment leakage. The leakage is minimized by continued Type B and C testing for penetrations which are in direct communication with the containment atmosphere. Also, the In-Service Inspection (ISI) program and maintenance rule program require periodic inspection of the interior and exterior of the containment structure to identify degradation. The results for the last two Type A tests are reported in the following table for Monticello:

Date	As-Found Leakage(*)	Acceptance Limit(**)	Test Pressure# (psia)
3/21/1993	0.6183 percent by weight per day	0.9000 percent by weight per day	56.538
11/11/1989	0.8240 percent by weight per day	0.9000 percent by weight per day	57.5592

* This is the leakage attributable to containment leakage as well as a number of Type B and Type C leakage components being tested as part of the Type A test.

** The total allowable "as-left" leakage is 0.75 L_a (L_a, 1.2% of primary containment air by weight per day, is the leakage assumed in dose consequences) with 0.6 L_a, the maximum leakage from Type B and C components.

\# The test pressure is the pressure recorded at the end of the test.

EXHIBIT A

Evaluation of Proposed Changes to the Monticello Technical Specifications

Benefits of the Proposed Change

The next Monticello ten-year Type A test is scheduled to be performed during Refueling Outage 21, which is currently scheduled to begin in April 2003. By allowing the one-time exception, NMC will:

- Perform the next Monticello Type A test no later than March 2008.

- Realize a substantial cost savings by not performing the Type A test for an additional five years. The estimated savings for the next Monticello outage include saving $300,000 associated with performance of the test, elimination of up to 48 hours of critical path outage time with associated replacement power cost savings of $500,000, and saving of personnel radiation exposure.

NMC understands that NEI is planning to seek NRC acceptance of a change to the NEI 94-01 guidance document with respect to Type A testing frequencies. It is anticipated that approval of the license amendment application will provide sufficient time for NEI to obtain NRC concurrence with the revised Type A testing frequency.

Determination of No Significant Hazards Considerations

Nuclear Management Company, LLC is requesting a revision to the Technical Specifications for the Monticello Nuclear Generating Plant, to incorporate a one-time exception to the ten-year frequency of the performance-based leakage rate testing program for Type A tests specified by Nuclear Energy Institute (NEI) 94-01 and endorsed by 10 CFR Part 50, Appendix J, Option B. This new exception will allow a Type A test to be performed within 15 years from the last Type A test for Monticello, which will require performance of the next Type A test by March 2008.

The proposed amendment has been evaluated to determine whether it constitutes a significant hazards consideration as required by 10 CFR Part 50, Section 50.91, using standards provided in Section 50.92(c). This analysis is provided below:

1. *The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed change to Technical Specification 4.7.A.2.b provides a one-time exception to the testing frequency for the Type A containment integrated leakage rate test. The current ten-year interval is based on past performance and the proposed change will only extend the Type A test frequency to fifteen years. The proposed change to the Technical Specifications does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary 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 primary containment does not involve the prevention or identification of any precursors of an accident and therefore does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of the evaluated accidents are the amount of radioactivity that is released to secondary containment and subsequently to the public. The proposed change involves a one-time change to the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency specified in the Monticello Technical Specifications. As documented in NUREG-1493, "Performance-Based Containment Leakage-Test Program," industry experience has shown that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment paths that are detected only by Type A tests is very small. An analysis of 144 integrated leak rate tests, including 23 failures, found that no failures were due to containment liner breach. NUREG-1493 also concluded, in part, that reducing the frequency of Type A containment leakage rate tests to once per twenty years was found to lead to an imperceptible increase in risk. The Monticello risk-based evaluation of the proposed one-time extension to the Type A test frequency supports this conclusion. The integrity of the reactor containment is subject to two types of failure mechanisms 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 primary containment, combined with the containment inspections performed in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI and 10 CFR 50.65, Maintenance Rule, provide a high degree of assurance that the primary containment will not degrade in a manner that is detectable only by Type A tests and therefore does not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.*

The proposed change to Technical Specification 4.7.A.2.b involves a one-time exception to the current test interval for Type A containment leakage rate tests. The primary containment and the test requirements invoked to periodically demonstrate the integrity of the primary containment exist to ensure the ability to mitigate the consequences of an accident. Additionally, the reactor containment and its associated test requirements do not involve the prevention or identification of any precursors of an accident. The proposed change to the leakage rate test frequency does not involve any physical changes being made to the facility. In addition, the proposed extension of the Type A leakage rate test frequency does not change the operation of the plant such that a new failure mode involving the possibility of a new or different kind of accident from any accident previously evaluated is created.

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

3. *The proposed amendment will not involve a significant reduction in the margin of safety.*

The proposed Technical Specification 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 proposed change involves only the extension of the interval between Type A containment leakage tests. The current interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. Type B and C containment leakage tests will continue to be performed at the frequency currently required by the plant Technical Specifications.

The NUREG-1493 generic study of the effects of extending containment leakage test intervals found that a twenty-year extension for Type A leakage tests resulted in an imperceptible increase in risk to the public. This study also found that, generically, the containment leakage paths are mainly detected by Type B and C tests. The proposed change involves a one-time extension of the frequency for Type A containment leakage tests; the overall primary containment leakage rate limit, specified by the Monticello Technical Specifications, is being maintained. The regular containment inspections being performed in accordance with the ASME Code, Section XI, and 10 CFR 50.65, Maintenance Rule, provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A tests. In addition, the containment monitoring capability that is inherent to boiling water reactors using an inert containment atmosphere allows for the detection of gross containment leakage that may develop during power operation. The cumulative effect of these inspections, tests and operating methods ensures that the margin of safety is maintained.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Base on the above evaluation, NMC has determined that the proposed amendment will not involve a significant hazards consideration.

EXHIBIT A

Evaluation of Proposed Changes to the Monticello Technical Specifications

Environmental Assessment

Nuclear Management Company, LLC has evaluated the proposed change and determined that:

1. The changes do not involve a significant hazards consideration.
2. The change does not involve a significant change in the type or significant increase in the amounts of any effluent that may be released offsite.
3. The change does not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51, Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51, Section 51.22(b), an environmental assessment of the proposed change is not required.

Risk Assessment for

Monticello Nuclear Generating Plant

Regarding

ILRT (Type A) Extension Request

RISK ASSESSMENT FOR

MONTICELLO NUCLEAR GENERATING PLANT

REGARDING

ILRT (TYPE A) EXTENSION REQUEST

Prepared for:

The Nuclear Management Company (NMC)
Monticello Nuclear Generating Plant

Prepared by:

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April 2002

MONTICELLO NUCLEAR GENERATING PLANT

RISK ASSESSMENT FOR THE MONTICELLO NUCLEAR GENERATING PLANT REGARDING THE ILRT (TYPE A) EXTENSION REQUEST

Prepared by: Donald E. Vanover Date: 4/5/02

Reviewed by: Donald E. MacLeod Date: 4/5/02

Approved by: [illegible] Date: 4/5/02

Accepted by: [illegible] Date: 4-8-02

Revisions:

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

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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 Monticello Nuclear Generating 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 [3], 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 [4].

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 (allowable leakage).

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 [5], "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.

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], 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 for a comparable BWR plant, that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on Monticello specific models and available data.

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 [3] builds on the EPRI Risk Assessment methodology, EPRI TR-104285 [2] (Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals). The NEI guidance 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 Regulatory Guide 1.174 [4] 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 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, which helps to ensure that the defense-in-depth philosophy is maintained, will also be calculated.

In addition, based on the precedent of other ILRT extension requests [6,18,20], the total annual risk (person rem/yr population dose) is examined to demonstrate the relative change in that parameter. (No threshold has 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 [3], EPRI TR-104285 [2] and NUREG-1493 [5]. The analysis uses the current Monticello PSA model that includes the results from the Monticello 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 frequency 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 [4] and compare with the acceptance guidelines of RG 1.174.

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

- Consistent with the other industry containment leak risk assessments, the Monticello assessment uses population dose as one of the risk measures. The other risk measures used in the Monticello 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 [3], the Monticello evaluation uses similar ground rules and methods to calculate the changes in risk metrics.

Section 3

GROUND RULES

The following ground rules are used in the analysis:

- The Monticello Level 1 and Level 2 internal events PSA model provides representative results for the analysis.
- It is appropriate to use the Monticello 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.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR 105189 [8].
- 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 Monticello compared to the NUREG/CR-4551 reference plant.
- The lowest consequence calculations (i.e., intact containment and small leakages) are based on scaling the NUREG/CR-4551 results for such cases using population differences, and also based on differences in the allowable Technical Specification Leakage.
- 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].

- Class 3b is conservatively categorized as LERF based on the previously approved methodology [6, 7].

- The impact on population doses from Interfacing System LOCAs 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 ISLOCA 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 utilized (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 [12]
4) NUREG/CR-4330 [13]
5) EPRI TR-105189 [8]
6) NUREG-1493 [5]
7) EPRI TR-104285 [2]
8) NUREG-1150 [14] and NUREG/CR-4551 [9]
9) NEI Interim Guidance from November 2001 [3]

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 eighth 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 Monticello. And the ninth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval that is followed in this analysis.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [15] 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 (PNL) 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. The study calculated unavailabilities for Technical Specification leakages and "large" leakages.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all of the 4 identified large leakage events were PWR events, and the assumption of a one-year duration is not applicable to an inerted containment such as Monticello. NUREG/CR-4220 identifies inerted BWRs as having significantly improved potential for leakage detection because of the requirement to remain inerted during power operation. This calculation presented in NUREG/CR-4220 is called an "upper bound" estimate for BWRs (presumably meaning "inerted" BWR containment designs).

NUREG-1273 [12]

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 [13]

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. For the BWR, the benefit from extending the ILRT frequency from 3 per 10 years to 1 per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

- Reduced opportunity for draindown events
- Reduced time spent in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS draindown events caused by ILRT/LLRT activities, and the other 5 were events involving loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the <u>improvement</u> in the safety due to extending the ILRT test interval.

<u>NUREG-1493 [5]</u>

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.

- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population 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 Integrated Leak Rate Test (ILRT) and (Local Leak Rate Test) 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 sequences 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 [14] and NUREG/CR 4551 [9]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Technical Specification leakage). This ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Peach Bottom. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the Monticello Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent Monticello if the Technical Specification leakage and the population are scaled to represent Monticello. (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 [3]

The guidance provided in this document builds on the EPRI Risk Impact Assessment methodology [2] and the NRC Performance-Based Containment Leakage Test Program [5], 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 Monticello 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 4.3 and 4.4.

4.2 PLANT SPECIFIC INPUTS

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

- Level 1 Model

- Level 2 Model
- 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]

The current Level 1 and Level 2 PSA model baseline results were used as the starting point for this analysis. The total release frequency including intact containment conditions from the Level 2 sequence quantification is 1.57E-05/yr. A breakdown of those results in several different categories from the Level 2 model is shown in Table 4.2-1 [17].

Population Dose Calculations

The population dose is calculated by using data provided in NUREG/CR-4551 and adjusting the results for Monticello. Each accident sequence 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 10 bins that are relevant to the analysis. Information from the Monticello PSA Containment Event Trees (CETs) was used to classify each of the Level 2 sequences using these attributes. The definitions of the 10 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 associated with each APB from NUREG/CR-4551.

(1) The two most recent Type A tests at Monticello (Surveillance Procedure 0136) have been successful, so the current Type A test interval requirement is 10 years [16].

Table 4.2-1
Monticello Level 2 PSA Model Plant Damage States [1]

RPV Pressure	R — Recovered in vessel	7.19E-06
	L — Vessel pressure low at lower head penetration	1.14E-06
	H — Vessel pressure high at lower head penetration	7.36E-06
	Total based on RPV Pressure Categorization:	**1.57E-05**
Containment Failure Mode	XX — Containment intact	7.39E-06
	VS — Containment vented through pool	1.67E-07
	VB — Containment vented bypassing pool	3.52E-06
	OD — Overpressure failure due to steam or non-condensable gas generation	1.83E-06
	OT — Over-temperature failure	7.64E-07
	OA — Overpressure failure due to steam generation from ATWS	5.62E-08
	OH — Overpressure failure due to hydrogen combustion	1.73E-06
	OE — Containment failure due to early severe accident challenges	1.33E-07
	LM — Liner melt through	0.00E+00
	CI — Containment isolation failure	2.95E-09
	BY — Containment bypass	9.33E-08
	Total Based on Containment Failure Mode Categorization:	**1.57E-05**
Timing of Release	X — Containment intact	7.39E-06
	L — Late release (~24 hrs)	5.45E-06
	I — Intermediate release (6-24 hrs)	7.64E-07
	E — Early release (< 6 hrs)	2.08E-06
	Total Based on Timing of Release Categorization:	**1.57E-05**

[1] Note that one sub-category from each of the major categories (i.e., RPV Pressure, Containment Failure Mode, and Timing of Release) is assigned to each of the Level 2 sequences. Hence, the total for each category is the same (i.e., 1.57E-05).

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

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.

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

Collapsed APB Number	Description
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

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

Collapsed Accident Progression (APB) Number	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 7.9 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	0.021	0.1659	9.55E-08	1.74E+06
2	0.0066	0.05214	4.77E-08	1.09E+06
3	0.556	4.3924	1.48E-06	2.97E+06
4	0.226	1.7854	7.94E-07	2.25E+06
5	0.0022	0.01738	1.30E-08	1.34E+06
6	0.059	0.4661	2.04E-07	2.28E+06
7	0.118	0.9322	4.77E-07	1.95E+06
8	0.0005	0.00395	7.99E-07	4.94E+03
9	0.01	0.079	3.86E-07	2.05E+05
10	0	0	4.34E-08	0
Totals	**1.0**	**7.9**	**4.34E-6**	

(1) Mean Fractional Contribution to Risk from Table 5.2-3 of NUREG/CR-4551

(2) The total population dose risk at 50 miles from internal events in person-rem is provided 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 2.5-6. 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.

Population Estimate Methodology

The person-rem results in Table 4.2-3 can be used as an approximation of the dose for Monticello if it is corrected for the population surrounding Monticello and the difference in Technical Specifications leak rate. For the updated population estimate, data is available for population by county from the US Census Bureau on the web site (http://quickfacts.census.gov/qfd/states/27000.html). This data is used to estimate the population within a 50-mile radius of the plant. If the entire county falls within the 50-mile radius based on a review of a map containing a mileage scale and county borders, then the entire population can be included in the population estimate. Otherwise, a fraction of the population is counted based on the percentage of the county within the 50-mile radius. The land area within the 50-mile radius is estimated based on visual inspection of the map and the population of that area is estimated assuming uniform distribution of the population within the county. The results of this updated population estimate are presented in Table 4.2-4.

Table 4.2-4
Population Within 50 Miles of Monticello (2000 US Census)

County Name	County Population		Population Within 50 Miles of Monticello
	Total	Percent Within 50 Miles of Monticello	
Anoka County	298,084	100	298,084
Benton County	34,226	100	34,226
Carver County	70,205	100	70,205
Chisago County	41,101	90	36,991
Dakota County	355,904	25	88,976
Hennepin County	1,116,200	100	1,116,200
Isanti County	31,287	100	31,287
Kanabec County	14,996	50	7,498
Kandiyohi County	41,203	5	2,060
McLeod County	34,898	80	27,918
Meeker County	22,644	100	22,644
Mille Lacs County	22,330	75	16,748
Morrison County	31,712	40	12,685
Pine County	26,530	10	2,653
Ramsey County	511,035	100	511,035
Renville County	17,154	2	343
Scott County	89,498	80	71,598
Sherburne County	64,417	100	64,417
Sibley County	15,356	20	3,071
Stearns County	133,166	70	93,216
Washington County	201,130	80	160,904
Wright County	89,986	100	89,986
		Total =	2,762,746

The total population shown above in Table 4.2-4 is compared to the total population 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 Monticello.

Total Monticello Population = 2.76E+06 [Table 4.2-4]
PBAPS Population from NUREG/CR-4551 = 3.02E+06 [18]
Population Dose Factor = 2.76E+06 / 3.02E+06 = 0.91

This population dose factor then can be applied to the APBs from NUREG/CR-4551. Additionally, a second correction factor is also required to be applied to the NUREG/CR-4551 calculation to account for differences in the Technical Specification [19] leakage value for Accident Progression Bin 8. The Technical Specification containment available leak rate for Monticello is 1.2% (L_a^M) versus the 0.5% (L_a^{PB}) for the NUREG-1150 plant, Peach Bottom. Therefore, the leakage (L_a^{PB}) person-rem calculated for Peach Bottom that is scaled by population for the Monticello analysis must be multiplied by a factor of 2.4 (L_a^M / L_a^{PB}) to account for the differences in Technical Specification leakage rates.

Table 4.2-5 shows the results of applying the population dose factor and the allowable leakage factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Monticello.

Table 4.2-5
Calculation of Monticello Population Dose Risk at 50 Miles

Accident Progression Bin #	NUREG/CR-4551 Population Dose at 50 miles (Person-rem)	Bin Multiplier used to obtain Monticello Population Dose	Monticello Adjusted Population Dose at 50 miles (Person-rem)
1	1.74E+06	0.91	1.58E+06
2	1.09E+06	0.91	9.92E+05
3	2.97E+06	0.91	2.70E+06
4	2.25E+06	0.91	2.05E+06
5	1.34E+06	0.91	1.22E+06
6	2.28E+06	0.91	2.07E+06
7	1.95E+06	0.91	1.77E+06
8	4.94E+03	2.4 x 0.91	1.08E+04
9	2.05E+05	0.91	1.87E+05
10	0	0	0.00E+00

Application of Monticello 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 Monticello 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 apply the Monticello PSA Level 2 model results into a format which allowed for the scaling of the Level 3 results based on current Level 2 output. Finally, as mentioned above, the Level 3 results were modified to reflect the difference in the site demographics that exist between the two sites. This subsection provides a description of the process used to apply the Monticello PSA Level 2 model results into a form that can be used to generate Level 3 results using the NUREG/CR-4551 documentation.

The basic process that was pursued to obtain Level 3 results based on the Monticello PSA Level 2 model and NUREG/CR-4551 was to define a useful relationship between the Level 2 and Level 3 results. Consequently, each sequence of the Monticello PSA Level 2 model was reviewed and assigned into one of the collapsed Accident Progression Bins (APBs) from NUREG/CR-4551. The Level 2 model contains a significantly larger amount of information about the accident sequences than what is used in the collapsed APBs in NUREG/CR-4551 and this assignment process required simplification of accident progression information and assumptions related to categorizations of certain items. Note that each Level 2 sequence is characterized by a combination of three Plant Damage State categories. The first characteristic is based on the RPV Pressure; the second characteristic is based on the Containment Failure Mode; and the third characteristic is based on the Timing of Release. The assumptions or rules used for the assignments to one of the NUREG/CR-4551 collapsed APBs for Monticello are shown in Table 4.2-6. With the assumptions listed in Table 4.2-6, it is possible to assign a representative APB from NUREG/CR-4551 to each of the Monticello Level 2 sequences.

Table 4.2-6
Monticello Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins

Category	Monticello Level 2 PSA Model Plant Damage States	Assumption
RPV Pressure	R - Recovered in vessel	All sequences (except those that were also characterized with "XX" for containment intact) that are characterized as recovered in vessel "R" are assigned to APB#9.
	L – Vessel pressure low at lower head penetration	The low pressure characteristic "L" is used for assignment to APB#2 for early wetwell failures or to APB#4 for early drywell failures. The RPV pressure is not a consideration in the late containment failure or containment intact sequences.
	H – Vessel pressure high at lower head penetration	The high pressure characteristic "H" is used for assignment to APB#1 for early wetwell failures or to APB#3 for early drywell failures. The RPV pressure is not a consideration in the late containment failure or containment intact sequences.

Table 4.2-6
Monticello Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins

Category	Monticello Level 2 PSA Model Plant Damage States	Assumption
Containment Failure Mode	XX — Containment intact	All containment intact sequences characterized with an "XX" plant damage state are assigned to APB#8.
	VS — Containment vented through pool	"VS" sequences with an early "E" release characteristic are assigned to APB#1 or APB#2 depending on the status of RPV pressure, "H" or "L", respectively. "VS" and "L" sequences are assigned to APB#5 unless the vessel status is "R" in which case, the sequence is assigned to APB#9 as described above. Note that this approach preferentially ignores APB#7 for containment vent scenarios since APB#7 does not distinguish between wetwell and drywell venting locations.
	VB — Containment vented bypassing pool	"VB" sequences with an early "E" release characteristic are assigned to APB#3 or APB#4 depending on the status of RPV pressure, "H" or "L", respectively. "VB" and "L" sequences are assigned to APB#6 unless the vessel status is "R" in which case, the sequence is assigned to APB#9 as described above. Note that this approach preferentially ignores APB#7 for containment vent scenarios since APB#7 does not distinguish between wetwell and drywell venting locations.
	OD — Overpressure failure due to steam or non-condensable gas generation	"OD" sequences are represented by late "L" drywell failures, and as such, are assigned to APB#6.
	OT — Over-temperature failure	"OT" sequences are represented by intermediate "I" drywell failures. These sequences are also assigned to APB#6 since the intermediate release timing is assumed to be closer to an "L" characteristic release than an "E" characteristic release.
	OA — Overpressure failure due to steam generation from ATWS	"OA" sequences are represented by early failures that could occur in the wetwell or drywell. The ATWS Level 2 Containment Event Tree was examined to provide more detailed binning assignments. Individual wetwell failure sequences were assigned to either APB#1 or APB#2 depending on the RPV pressure, "H" or "L", respectively. Similarly, the drywell failure sequences were assigned to APB#3 or APB#4 depending on the RPV pressure. "OA" sequences that also included a recovered, "R" PDS value, were assigned to APB#9 consistent with all "R" sequences.
	OH — Overpressure failure due to hydrogen combustion	"OH" sequences are represented by early "E" drywell failures, and are assigned to APB#3 or APB#4 depending on the status of the RPV pressure, "H" or "L", respectively.
	OE — Containment failure due to early severe accident challenges	"OE" sequences are represented by early "E" drywell failures, and are assigned to APB#3 or APB#4 depending on the status of the RPV pressure, "H" or "L", respectively.

Table 4.2-6
Monticello Level 2 Model Assumptions for Application to the NUREG/CR-4551 Accident Progression Bins

Category	Monticello Level 2 PSA Model Plant Damage States	Assumption
Containment Failure Mode	LM — Liner melt through	"LM" sequences are represented by early "E" drywell failures, and are assigned to APB#3 or APB#4 depending on the status of the RPV pressure, "H" or "L", respectively.
	CI — Containment isolation failure	Containment isolation failure sequences "CI" could be assigned to APB#3 or APB#4, but in the EPRI methodology are called out separately and are assigned directly to EPRI Class 2.
	BY — Containment bypass	Containment bypass sequences "BY" could also be assigned to APB#3 or APB#4, but in the EPRI methodology are called out separately and are assigned directly to EPRI Class 8.
Timing of Release	X — Containment intact	"X" sequences always appear in combination with an "XX" characteristic, and as such, are always assigned to APB#8.
	L — Late release (~24 hrs)	For non-recovered sequences, i.e. not equal to "R", an "L" sequence characteristic resulted in an assignment to APB#5 or APB#6 depending on the expected containment failure location in the wetwell or drywell, respectively.
	I — Intermediate release (6-24 hrs)	The only "I" sequences were combined with "OT" failures, and as such were assigned to APB#6 as described above.
	E — Early release (< 6 hrs)	For non-recovered sequences, i.e., not equal to "R", an "E" sequence characteristic resulted in an assignment to APB#1, APB#2, APB#3, or APB#4 depending on the expected containment failure location in the wetwell or drywell, respectively, and on the status of the RPV pressure as described above.

Release Category Definitions

Table 4.2-7 defines the accident classes used in the ILRT extension evaluation consistent with the EPRI methodology [2].

Table 4.2-7
EPRI 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 [3]. 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).

4.4 IMPACT OF EXTENSION ON LEAK DETECTION PROBABILITY

Consistent with the NEI Guidance [3], 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)[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.

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(2) These are obviously approximations assumed by the NRC and EPRI because the 3 ILRTs in 10 year requirement would have a T/2 = 1.67 years instead of 1.5 years.

Section 5
RESULTS

The application of the approach based on NEI Interim Guidance [3], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 18, 20] have led to the following results. The method chosen to display the results is according to the eight (8) accident classes consistent with these previous evaluations. Table 5-1 lists these accident classes.

The analysis performed examined Monticello 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).
- Small containment isolation "failure-to-seal" events (EPRI TR-104285 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. Consistent with the NEI Guidance, these Classes are not specifically examined since they will not significantly influence the results of this analysis.
- 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 also 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), and large containment isolation failures (EPRI TR-104285 Class 2 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.

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

The severe accident sequence frequencies that can result in offsite consequences are evaluated. The latest update of the Monticello Level 2 PSA model is used in the ILRT evaluation [17].

This step involves the review of the Monticello containment event trees (CETs) and Level 2 accident sequence frequency results. The CETs characterize the response of the containment to important severe accident sequences. As described in Section 4.2, each of the Monticello Level 2 sequences were examined and each endstate was applied to one of the Accident Progression Bins as defined in NUREG/CR-4551. This application forms the basis for estimating the population dose for Monticello.

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 analysis. (These events are represented by the "Class 3" sequence depicted in EPRI TR-104285 [2]). The question on containment status is 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 are then considered in addition to large containment failure modes. These are Event CLASS-3a (small breach) and Event CLASS-3b (large breach).

After including the respective "small" and "large" liner breach leak rate probabilities, the eight severe accidents class frequencies are 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 Containment Failure Mode Category "XX" from table 4.2-1 minus the EPRI/NEI Class 3a and 3b frequency.

Class 2 Sequences. 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 Level 2 Containment Failure Mode Category "CI" from Table 4.2-1. The value of 2.95E-9 was determined from the sum of all Level 2 sequences 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]

These probabilities are conservatively multiplied by the total CDF value to obtain a first order estimate of the Class 3a and Class 3b frequencies. Additionally, the dose associated with containment leakage for Class 3a is $10L_a$ and for Class 3b, it is $35L_a$. These frequency and dose assignments are consistent with the guidance prescribed in EPRI/NEI Interim Guidance [3].

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. Consistent with the NEI interim guidance [3], since 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. Consistent with the NEI interim guidance [3], since these 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. Also consistent with the NEI interim guidance [3], 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., over-pressure or over-temperature). For this analysis, the associated radionuclide releases are based on the application of the Level 2 end states to the Accident Progression Bins from NUREG/CR-4551 as described in Section 4.2. The Class 7 Sequences are divided into 8 categories that can be mapped directly to Bins 1-7, and 9 from NUREG/CR-4551. The failure frequency and population dose for each specific APB is shown below in Table 5-2. The total release frequency and total dose are then used to determine a weighted average person-rem for use as the representative EPRI Class 7 dose in the subsequent analysis. Note that the total frequency and dose associated from this EPRI class does not change as part of the ILRT extension request.

Table 5-2
ACCIDENT CLASS 7 FAILURE FREQUENCIES AND POPULATION DOSES (MONTICELLO BASE CASE LEVEL 2 MODEL)

Accident Class (APB umber)	Release Frequency/yr	Population Dose (50 miles) Person-Rem (1)	Population Dose Risk (50 Miles) (Person-Rem/yr) (2)
7a (APB #1)	9.87E-08	1.58E+06	1.56E-01
7b (APB #2)	1.70E-07	9.92E+05	1.69E-01
7c (APB #3)	2.24E-07	2.70E+06	6.05E-01
7d (APB #4)	1.30E-07	2.05E+06	2.66E-01
7e (APB #5)	1.53E-09	1.22E+06	1.87E-03
7f (APB #6)	6.11E-06	2.07E+06	1.27E+01
7g (APB #7)	0.00E+00	1.77E+06	0.00E+00
7h (APB #9)	1.46E-06	1.87E+05	2.73E-01
Class 7 Total	**8.20E-06**	**1.73E+06** (3)	**14.15E+00**

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(1) Population dose values obtained from Table 4.2-5 based on the Accident Progression Bin.
(2) Obtained by multiplying the Release Frequency value from the second column of this table by the Population dose value from the third column of this table.
(3) The weighted average population dose for Class 7 is obtained by dividing the total population dose risk by the total release frequency.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. The containment bypass failure frequency is obtained from the Level 2 Containment Failure Mode Category "BY" from Table 4.2-1. From the base Level 2 model results, this frequency is 9.33E-8/yr.

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. Table 5-3 summarizes these accident frequencies by Accident Class.

Table 5-3
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (MONTICELLO BASE CASE)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)
1	No Containment Failure	6.92E-06
2	Large Isolation Failures (Failure to Close)	2.95E-09
3a	Small Isolation Failures (liner breach)	4.23E-07
3b	Large Isolation Failures (liner breach)	4.23E-08
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)	8.20E-06
8	Bypass (Interfacing System LOCA)	9.33E-08
CDF	**All CET End states (including very low and no release)**	**1.57E-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 and allowable leakage compared to the reference plant as described in Section 4.2, and summarized in Table 4.2-5. The results of applying these releases to the EPRI containment failure classification are shown below.

Class 1	=	1.08E4 person-rem (at 1.0L_a)	= 1.08E4 person-rem (1)
Class 2	=	2.70E6 person-rem (2)	
Class 3a	=	1.08E4 person-rem x 10L_a	= 1.08E5 person-rem (3)
Class 3b	=	1.08E4 person-rem x 35L_a	= 3.78E5 person-rem (3)
Class 4	=	Not analyzed	
Class 5	=	Not analyzed	
Class 6	=	Not analyzed	
Class 7	=	1.73E6 person-rem (4)	
Class 8	=	2.70E6 person-rem (5)	

(1) The population dose associated with the Technical Specification Leakage is based on scaling both the population data and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant. The derivation is described in Section 4.2 for Monticello. The release for this Class is assigned from APB#8 from table 4.2-5.

(2) Class 2 (Containment Isolation failures) may be drywell isolation failures. Therefore, the release associated with this Class is assigned to be equivalent to the releae associated with APB#3 from Table 4.2-5.

(3) The population dose for Technical Specification Leakage is derived as discussed in Note (1) and the Class 3a and 3b releases are related to the Technical Specification Leakage rate as shown. This is consistent with previous submittals [6, 20], and the NEI guidance [3].

(4) This is the weighted average person-rem for Class 7 as derived in Table 5-2 for APBs #1-7 and 9.

(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 assumed to result in a direct path to the environment, and as such, are assigned to be equivalent to the release associated with APB#3 from Table 4.2-5.

The population dose estimates derived for use in the risk evaluation per the EPRI methodology containment failure classification are summarized in Table 5-4.

Table 5-4
MONTICELLO POPULATION DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	1.08E+04
2	Large Isolation Failures (Failure to Close)	2.70E+06
3a	Small Isolation Failures (liner breach)	1.08E+05
3b	Large Isolation Failures (liner breach)	3.78E+05
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+06
8	Bypass (Interfacing System LOCA)	2.70E+06

The above results when combined with the results presented in Table 5-3 yield the Monticello baseline mean consequence measures for each accident class. These results are presented in Table 5-5 below:

Table 5-5
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS (I.E., REPRESENTATIVE OF THE INITIAL ILRT DATA SET)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure [(1)]	6.92E-06	1.08E+04	7.47E-02
2	Large Isolation Failures (Failure to Close)	2.95E-09	2.70E+06	7.97E-03
3a	Small Isolation Failures (liner breach)	4.23E-07	1.08E+05	4.57E-02
3b	Large Isolation Failures (liner breach)	4.23E-08	3.78E+05	1.60E-02
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)	8.20E-06	1.73E+06	1.42E+01
8	Bypass (Interfacing System LOCA)	9.33E-08	2.70E+06	2.52E-01
CDF	**All CET End states (including very low and no release)**	**1.57E-05**		**14.55**

(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 total dose is comparable with the other sites as shown below:

Plant	Annual Dose (Person-Rem/Yr)	Reference
Indian Point 3	14,515	[6]
Monticello	14.6	[Table 5-5]
Peach Bottom	6.2	[18]
Crystal River	1.4	[20]

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

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

Where:

$CLASS3a_{BASE}$ = Class 3a person-rem/year = 4.57E-2 person-rem/year [Table 5-5]

$CLASS3b_{BASE}$ = Class 3b person-rem/year = 1.60E-2 person-rem/year [Table 5-5]

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 14.55 person-rem/yr [Table 5-5]

$$\%Risk_{BASE} = [(4.57E\text{-}2 + 1.60E\text{-}2) / 14.55] \times 100 = (6.17E\text{-}2) / 14.55$$

$$\%Risk_{BASE} = 0.4\%$$

Therefore, the Total Type A 3/10-years ILRT interval risk contribution of leakage, represented by Class 3a and Class 3b accident scenarios is 0.4% for Monticello.

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 5-6.

Table 5-6
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	5.84E-06	1.08E+04	6.30E-02
2	Large Isolation Failures (Failure to Close)	2.95E-09	2.70E+06	7.97E-03
3a	Small Isolation Failures (liner breach)	1.41E-06	1.08E+05	1.52E-01
3b	Large Isolation Failures (liner breach)	1.41E-07	3.78E+05	5.33E-02
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)	8.20E-06	1.73E+06	1.42E+01
8	Bypass (Interfacing System LOCA)	9.33E-08	2.70E+06	2.52E-01
CDF	**All CET End states (including very low and no release)**	**1.57E-05**		**14.68**

Based on the risk values from Tables 5-6, the percent risk contribution ($\%Risk_{10}$) for Class 3 is as follows:

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

Where:

$CLASS3a_{10}$ = 1.52E-1 person-rem/year [Table 5-6]

$CLASS3b_{10}$ = 5.33E-2 person-rem/year [Table 5-6]

$TOTAL_{10}$ = 14.68 person-rem/yr [Table 5-6]

$\%Risk_{10}$ = [(1.52E-1 + 5.33E-2) / 14.68] *100 = (2.05E-1) / 14.68

$\%Risk_{10}$ = 1.4%

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3a and Class 3b accident scenarios is 1.4%.

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 5-7.

Table 5-7
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	5.06E-06	1.08E+04	5.46E-02
2	Large Isolation Failures (Failure to Close)	2.95E-09	2.70E+06	7.97E-03
3a	Small Isolation Failures (liner breach)	2.12E-06	1.08E+05	2.28E-01
3b	Large Isolation Failures (liner breach)	2.12E-07	3.78E+05	8.00E-02
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)	8.20E-06	1.73E+06	1.42E+01
8	Bypass (Interfacing System LOCA)	9.33E-08	2.70E+06	2.52E-01
CDF	**All CET End states (including very low and no release)**	**1.57E-05**		**14.77**

Based on the values from Table 5-7, 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:

$CLASS3a_{15}$ = 2.28E-1 person-rem/year [Table 5-7]

$CLASS3b_{15}$ = 8.00E-2 person-rem/year [Table 5-7]

$TOTAL_{15}$ = 14.77 person-rem/yr [Table 5-7]

$\%Risk_{15}$ = [(2.28E-1 + 8.00E-2) / 14.77] *100 = (3.08E-1) / 14.77

$\%Risk_{15}$ = 2.1%

Therefore, the Total Type A 15-year ILRT interval risk contribution of leakage, represented by Class 3a and Class 3b accident scenarios is 2.1%.

In summary, the results above show that the percent contribution from risk due to ILRT-averted leakage scenarios is small 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 <u>total integrated plant risk</u> when the ILRT is extended from 10 years to 15 years is computed as follows:

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

Where:

$TOTAL_{10}$ = 14.68 person-rem/year [Table 5-6]

$TOTAL_{15}$ = 14.77 person-rem/year [Table 5-7]

$\%TOTAL_{10\text{-}15}$ = [(14.77 – 14.68) / 14.68] x 100

$\%TOTAL_{10\text{-}15}$ = 0.6%

Therefore, the risk impact on the total integrated plant risk for an ILRT extension from 10 to 15 years is only 0.6% for Monticello.

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. The Class 3b radionuclide release person-rem is significantly less than a typical LERF contributor as seen by comparing the relative population dose for Class 3b/Class 2 (3.78E5 person-rem / 2.70E6 person-rem) or 14%. Nevertheless, Class 3b is treated in this analysis as a potential LERF contributor. Class 3a is even less than Class 3b and is treated here as not a "large" release. Therefore, for this evaluation, only Class 3b sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Other accident classes such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval.

Late releases are excluded regardless of the size of the leak because late releases are, by definition, not LERF contributors. (See also the discussion in Section 5.5 regarding the conditional containment failure probability to assess the defense-in-depth.) Therefore, the frequency of Class 3b sequences is used as a conservative surrogate for estimating the change in LERF. This frequency, based on a three-year test interval, is 4.23E-8/yr [Table 5-5]; based on a ten-year test interval, it is 1.41E-7 [Table 5-6]; and, based on a fifteen-year test interval, it is 2.12E-7 [Table 5-7]. Thus, increasing the ILRT test interval from 10 to 15 years results in an additional 7.1E-8/yr increase in the overall LERF value as measured by the increase in Class 3b sequences. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1E-7/yr. Therefore, using this NRC guidance, increasing the ILRT interval to 15 years represents a very small change in risk.

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, based on the NEI guidance [3] methodology, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state (Class 1) and small failures (Class 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Consequently, the change in CCFP can be calculated by the difference in the Class 1 and Class 3a frequencies:

$$CCFP_{\%} = 1 - (\text{Intact Containment Frequency / Total CDF}) \times 100\%, \text{ or}$$
$$= 1 - [(\text{Class 1 Frequency + Class 3a Frequency) / CDF}] \times 100\%$$

So for a 10-year interval:

$$CCFP_{10} = 1 - [(5.84E\text{-}6 + 1.41E\text{-}6) / 1.57E\text{-}5] \times 100\%$$
$$= 53.8\%$$

And for a 15-year interval:

$$CCFP_{15} = 1 - [(5.06E\text{-}6 + 2.12E\text{-}6) / 1.57E\text{-}5] \times 100\%$$
$$= 54.2\%$$

Therefore, the change in the conditional containment failure probability is given by:

$$\Delta CCFP = CCFP_{15} - CCFP_{10} = 0.4\%$$

This change in CCFP of less than 1% 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. The baseline risk contribution (person-rem) of leakage, represented by Class 3 accident scenarios is 0.4% where the majority of the risk is associated with severe accident phenomena during core melt progression.
2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem) represented by Class 3 accident scenarios is 1.4%.
3. When the ILRT interval is 15 years, the risk contribution of leakage represented by Class 3 accident scenarios is 2.1%.
4. The total integrated increase in risk contribution from extending the ILRT test frequency from the current once-per-10 year interval to once-per-15 years is 0.6%.
5. The risk increase in LERF from extending the ILRT test frequency from the current once-per-10 year interval to once-per-15 years is 7.1E-8/yr. This is determined to be very small using the acceptance guidelines of Reg. Guide 1.174.
6. The change in the conditional containment failure frequency from the current once-per-10 year interval to once-per-15 years is 0.4%. Though no official acceptance criteria exists for this risk metric, it is also judged to be very small.
7. Other salient results are summarized in Table 5-8.

Table 5-8
SUMMARY OF RISK IMPACT ON TYPE A ILRT TEST FREQUENCY

Class [(1)]	Risk Impact (Base) [(2)]	Risk Impact (10-years) [(3)]	Risk Impact (15-years) [(4)]
3a and 3b	4.66E-07 / year 0.062 person-rem / year	1.55E-06 / year 0.206 person-rem / year	2.33E-06 / year 0.308 person-rem / year
3b (~LERF)	4.23E-08 / year 0.016 person-rem / year	1.41E-07 / year 0.053 person-rem / year	2.12E-07 / year 0.080 person-rem / year
Total Integrated Risk	1.57E-05 / year 14.55 person-rem / year	1.57E-05 / year 14.68 person-rem / year	1.57E-05 /year 14.77 person-rem / year
CCFP[(5)] (Total – Class 1 - Class 3a)	8.34E-06 / year 53.16%	8.44E-06 / year 53.79%	8.51E-06 / year 54.24%

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

(2) Monticello baseline values.

(3) Type A ILRT test interval of 1 in 10 years.

(4) Type A ILRT test interval of 1 in 15 years.

(5) Consistent with the NEI methodology [3], Class 1 and Class 3a represent containment "intact" conditions for determining the conditional containment failure probability.

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 [4] 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 using the change in the Class 3b frequency as a conservative surrogate for LERF is 7.1E-8. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the Monticello risk profile.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the total integrated plant risk for those accident sequences influenced by Type A testing by only 0.6%. Therefore, the risk impact when compared to other severe accident risks is negligible.

Risk Trade-Off

The performance of an ILRT introduces risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real risk impacts associated with the setup and performance of the ILRT during shutdown operation [8]. While these risks have not been quantified for Monticello, it is judged that there is a positive (yet unquantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT extension, there are in fact some positive safety benefits.

Previous Assessments

The NRC in NUREG-1493 [5] 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 failure.

The findings for Monticello confirm the above general findings on a plant specific basis when considering (1) the severe accidents evaluated for Monticello, (2) the Monticello containment failure modes, and (3) the local population surrounding Monticello.

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

[4] *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.

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

[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: Peach Bottom, Unit 2,* Main Report NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, December 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] *Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check',* NUREG-1273, April 1988.

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

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

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

[16] *Integrated Primary Containment Leak Rate Test*, Monticello Surveillance Procedure 0136, was performed with acceptable results on 11/5/1989 and 3/21/93. [Personal communication from A. Sarrack (NMC) to D. Vanover (ERIN)] March 21, 2002.

[17] *Level 2 Accident Sequence Quantification (Update 2A)*, NMC Calculation II.SMR.97.012, December 2001.

[18] 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

[19] *Monticello Nucelar Generating Plant Technical Specifications, Unit 1*, Docket Number 50-263, License Number DPR-22. LCO 3.7 (Containment Systems), item 3.7.A.2.b.1, Amendment 95, dated 4/3/96.

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

Exhibit C

License Amendment Request for

Risk-Informed Technical Specification Change

Regarding Five Year Extension of Type A Test Interval

Current Monticello Technical Specification Pages
Marked Up With Proposed Change

This Exhibit consist of current Monticello Technical Specification pages marked up with the proposed changes. The pages included in the exhibit are listed below:

TS Pages

159
185

b. When Primary Containment Integrity is required, leakage rates shall be limited to:

1. An overall integrated leakage rate of less than or equal to La, 1.2 percent by weight of the containment air per 24 hours at Pa, 42 psig.

2. A combined maximum flow path leakage rate of less than or equal to 0.6La for all penetrations and valves, subject to Type B and C tests when pressurized to Pa, 42 psig.

3. Less than or equal to 46 scf per hour combined maximum flow path leakage for all main steam isolation valves when tested at 25 psig.

With the measured overall integrated primary containment leakage rate exceeding 0.75La, or the measured combined leakage rate for all penetrations and valves subject to Type B and C testing exceeding 0.6La, or the measured combined maximum flow path leakage rate exceeding 46 scf per hour for all main steam isolation valves, restore leakage rates to less than or equal to these values prior to increasing reactor coolant system temperature above 212�F or, alternatively, restore measured leakage rates to within these limits within one hour or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

b. Perform required visual examinations and leakage rate testing for Type A containment integrated leakage rate tests in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and Regulatory Guide 1.163 dated September 1995**, as modified by the following exception:**

NEI 94-01 – 1995, Section 9.2.3:
The first Type A test performed after the March 1993 Type A test shall be performed no later than March 2008.

Perform Type B and C tests in accordance with 10 CFR 50,Appendix J, Option A, as modified by approved exemptions.

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Bases 4.7 (Continued):

On September 26, 1995, Regulatory Guide 1.163 became effective providing guidance on performance based testing to the requirements of 10 CFR 50, Appendix J, Option B. Monticello has adopted Option B, Section III.A of 10 CFR Part 50, Appendix J, for Type A primary reactor containment integrated leakage rate testing **as modified by the following exception: NEI 94-01 – 1995, Section 9.2.3: The first Type A test performed after the March 1993 Type A test shall be performed no later than March 2008**. Monticello will continue to perform Type B and C testing in accordance with 10 CFR Part 50, Appendix J, Option A.

NEXT PAGE IS PAGE 188

Exhibit D

License Amendment Request for

Risk-Informed Technical Specification Change

Regarding Five Year Extension of Type A Test Interval

Revised Monticello Technical Specification Pages

This Exhibit consist of revised Monticello Technical Specification Pages that incorporate the proposed changes. The Pages included in the exhibit are listed below:

TS Pages

159
185

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

b. When Primary Containment Integrity is required, leakage rates shall be limited to:

1. An overall integrated leakage rate of less than or equal to La, 1.2 percent by weight of the containment air per 24 hours at Pa, 42 psig.
2. A combined maximum flow path leakage rate of less than or equal to 0.6La for all penetrations and valves, subject to Type B and C tests when pressurized to Pa, 42 psig.
3. Less than or equal to 46 scf per hour combined maximum flow path leakage for all main steam isolation valves when tested at 25 psig.

With the measured overall integrated primary containment leakage rate exceeding 0.75La, or the measured combined leakage rate for all penetrations and valves subject to Type B and C testing exceeding 0.6La, or the measured combined maximum flow path leakage rate exceeding 46 scf per hour for all main steam isolation valves, restore leakage rates to less than or equal to these values prior to increasing reactor coolant system temperature above 212°F or, alternatively, restore measured leakage rates to within these limits within one hour or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

b. Perform required visual examinations and leakage rate testing for Type A containment integrated leakage rate tests in accordance with 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and Regulatory Guide 1.163 dated September 1995, as modified by the following exception:

NEI 94-01 - 1995, Section 9.2.3:
The first Type A test performed after the March 1993 Type A test shall be performed no later than March 2008.

Perform Type B and C tests in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.

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Bases 4.7 (Continued):

On September 26, 1995, Regulatory Guide 1.163 became effective providing guidance on performance based testing to the requirements of 10 CFR 50, Appendix J, Option B. Monticello has adopted Option B, Section III.A of 10 CFR Part 50, Appendix J, for Type A primary reactor containment integrated leakage rate testing as modified by the folllowing exceptions: NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the March 1993 Type A test shall be performed no later than March 2008. Monticello will continue to perform Type B and C testing in accordance with 10 CFR Part 50, Appendix J, Option A.

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