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A Duke Energy Company

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February 21, 2007

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke)
McGuire Nuclear Station, Unit 1
Docket Number 50-369
Proposed Technical Specification (TS) Amendment
TS 5.5.2 Containment Leak Rate Testing Program

References: Letters from Duke Energy Corporation to NRC, dated May 29, 2002, September 25, 2002, November 12, 2002, January 8, 2003, and January 29, 2003

Letter from NRC to Duke Energy Corporation, dated March 12, 2003

In accordance with the provisions of 10 CFR 50.90 and 10 CFR 50.4, Duke is requesting an amendment to the McGuire Facility Operating License and TS. This License Amendment Request (LAR) is to allow, on a one-time basis, an extension of the interval governing the conduct of the Integrated Leak Rate Test (ILRT). The proposed LAR revises administrative Technical Specification 5.5.2 "Containment Leak Rate Testing Program" from the currently approved 15-year interval (since the last McGuire Unit 1 Type A test) to a frequency encompassing the end of the McGuire Unit 1 End of Cycle (EOC) 19 refueling outage (approximately 6 months beyond the present frequency). As discussed in this LAR, the last McGuire Unit 1 ILRT was performed on May 27, 1993. The next ILRT is required, by TS 5.5.2, to be performed no later than May 26, 2008, approximately six months prior to the conclusion of refueling outage 1EOC19.

The above referenced May 29, 2002 letter requested a one-time extension of the interval governing the conduct of the ILRT from a 10 year frequency to a 15 year frequency. In addition, the September 25, 2002, November 12, 2002, January 8, 2003, and January 29, 2003 letters provided additional information as requested by the NRC.

A017

The March 12, 2003 NRC letter issued Amendment No. 211 to Facility Operating License NPF-9 to extend the ILRT interval period to 15 years.

The contents of this LAR follow the guidance in NEI 06-02 and contain a summary description, a detailed description, a technical evaluation, a regulatory evaluation including a no significant hazard consideration determination and an environmental consideration. As discussed in this LAR, the proposed change does not involve a significant hazard consideration pursuant to 10 CFR 50.92 and pursuant to 10 CFR 51.22 (c) (9) an Environmental Assessment / Impact Statement can be categorically excluded.

Amendment implementation will be accomplished within 30 days of NRC approval.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this LAR has been reviewed and approved by the McGuire Plant Operations Review Committee and the Duke Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this LAR is being sent to the appropriate state officials.

This letter and attachments do not contain any new NRC commitments.

Inquiries on this matter should be directed to K.L. Ashe at (704) 875-5715.

Sincerely,

Gary R. Peterson

Attachments

Gary R. Peterson affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

Gary R. Peterson

Gary R. Peterson, Vice President

Subscribed and sworn to me: February 21, 2007
Date

Deana A. DeLoach / Deana A. DeLoach
Notary Public

My commission expires: June 18, 2008
Date

SEAL

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ATTACHMENT 1

MARKED-UP TS FOR MCGUIRE

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Station Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

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

NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 27, 1993 (Unit 1) and August 20, 1993 (Unit 2) Type A test shall be performed no later than ~~May 26, 2008~~ **plant restart after the End Of Cycle 19 Refueling Outage** (Unit 1) and August 19, 2008 (Unit 2), and

ATTACHMENT 2

REPRINTED TS FOR MCGUIRE (TO BE PROVIDED TO NRC FOLLOWING COMPLETION OF TECHNICAL REVIEW)

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b. Shall become effective after the approval of the Station Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

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NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 27, 1993 (Unit 1) and August 20, 1993 (Unit 2) Type A test shall be performed no later than plant restart after the End Of Cycle 19 Refueling Outage (Unit 1) and August 19, 2008 (Unit 2), and

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Duke is requesting an amendment to the McGuire Nuclear Station Unit 1 Facility Operating Licenses and Technical Specifications (TS). This amendment will allow a one-time, approximately six month, extension to the currently approved 15-year test interval for the containment integrated leak rate test (ILRT). The McGuire Nuclear Station Unit 1 ILRT was last performed on May 27, 1993 and TS 5.5.2 currently requires that the next test be performed no later than May 26, 2008. This change proposes to extend the completion date for the next ILRT, approximately 6 months, thru the Unit 1 End of Cycle 19 outage. Extending the completion date prevents a premature plant shutdown to perform the test and provides time to plan for the ILRT in refueling outage 19 in the Fall of 2008.

2.0 DETAILED DESCRIPTION

2.1 Proposed Change

TS 5.5.2 currently includes the following:

<u>Containment Leakage Rate Testing Program</u>

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:
>
> a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 27, 1993 (Unit 1) and August 20, 1993 (Unit 2) Type A test shall be performed no later than May 26, 2008 (Unit 1) and August 19, 2008 (Unit 2), and

This proposed change modifies TS 5.5.2 to the following:

> a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 27, 1993 (Unit 1) and August 20, 1993 (Unit 2) Type A test shall be performed no later than plant restart after the End Of Cycle 19 Refueling Outage (Unit 1) and August 19, 2008 (Unit 2), and

2.2 Background Information

The testing requirements of 10 CFR Part 50, Appendix J provide assurance that the primary containment, including those systems and components that penetrate the primary containment, do not exceed the leakage rate assumed in the plant analyses. The main purpose of the reactor containment system is to mitigate the consequences of potential accidents by minimizing the release of radionuclides to the environment to assure the health and safety of the public. Appendix J specifies containment leakage testing requirements, which include an ILRT (also known as a Type A test). The ILRT measures the overall leakage rate of the primary containment.

McGuire TS 5.5.2 establishes the requirements for implementing a program to perform containment leakage rate testing in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J. The types of containment leakage tests include Type A (containment ILRT), Type B (local leak rate testing for containment penetrations, hatches, personnel air locks, etc.), and Type C (local leak rate testing for containment isolation valves). McGuire presently conducts Type A, Type B and Type C testing according to Option B.

10 CFR 50 Appendix J was revised in 1995 to allow use of Option B, Performance Based Requirements. Regulatory Guide (RG) 1.163 (Reference 1) concludes that NEI 94-01, Revision 0, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, provides methods acceptable to the NRC staff for complying with the provisions of Option B. Also, NEI 94-01 (Reference 2) permits an extended ILRT test interval of 10 years based on two consecutive successful tests. Successful ILRTs were completed on McGuire Unit 1 on May 2, 1990 and May 30, 1993. In March 2003, the NRC approved Amendment 211 (Reference 3) following Duke Power Company's request for a one time, five-year extension to the 10-year ILRT interval. The safety evaluation of this change, by the NRC, determined that the risk associated with the ILRT extension is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 (Reference 4). As a result, the McGuire Unit 1 due date for the next ILRT was extended to May 26, 2008.

This LAR proposes a one-time extension of the interval for the containment ILRT from 15 years to approximately 15.5 years. Extending the ILRT due date from May 26, 2008 to no later than plant restart after the End Of Cycle 19 Refueling Outage (Unit 1), would reduce concerns associated with incorporating the ILRT into the End of Cycle 18 refueling outage, prevent a forced outage, and provide time to plan and incorporate the containment ILRT in refueling outage 19 in the Fall of 2008. Refueling outage 19 is currently scheduled to end approximately six months after the current ILRT due date of May 26, 2008. Without the requested extension, McGuire Nuclear Station would be required to enter a forced outage to perform the test. Whereas, including the ILRT in refueling outage 18, which is scheduled for the spring of 2007 (14 months prior to the ILRT due date) could impact the overall length of the outage. The containment projects included in the scope of refueling outage 18 (e.g., Alloy 600 work, Cable Routing for the Digital Controls Upgrade, and Containment Sump modifications) would complicate the performance of the ILRT.

In addition, TS SR 3.6.1.1 Note 2 states that following each Type A test, leaking penetration bellows assemblies shall be subjected to a local leak test pressurized from the containment side of the assemblies at P_a. Although further degradation of penetration bellows is not expected, leaking bellows would require installation of Mechanical Inflatable Re-Useable Annular Penetrations Seal (MIRAPS) in order to perform the required leak tests. Many of the MIRAPS would have to be installed in the pipe chase and would involve a substantial amount of setup to install the MIRAPS after depressurization of the ILRT to complete the local bellows leak tests at P_a. Some are directly behind the sump penetrations and would directly conflict with work in the sump area. Most would involve transport of equipment and personnel past the sump penetrations which would interrupt work on the sump which will be on-going following the window for ILRT and subsequent depressurization. Although McGuire is only required to test the bellows following an ILRT, a more conservative approach (test plan) has been implemented. McGuire has developed and is presently utilizing a supplementary testing program that tests one-third of the bellows each outage. Under this program all bellows have now been tested.

The balance of listed work would involve conflicting labor resources for planning, work execution, inspection and testing as well as removal and replacement of welding tools and other utilities in containment as required by the ILRT.

3.0 TECHNICAL EVALUATION

The proposed change to extend the ILRT surveillance interval thru the end of Cycle 19 refueling outage (approximately six months) is justified based on the results of previous ILRTs, containment inspection programs, and a risk evaluation.

3.1 Previous ILRT Results

Previous ILRT testing confirmed that the McGuire Station Unit 1 containment structure leakage is acceptable, with considerable margin, with respect to the TS acceptance criterion of 0.30% of primary containment air weight per day (1.0 La). The test results and methods used to determine containment leakage was presented in the May 29, 2002 letter from Duke to the NRC (Reference 5) and for convenience is shown in Table 1 below.

Table - 1 Summary of Type A Test Results for McGuire Unit 1

Test Type	Test Date	Test Method	Test Results [1] (weight-percent/day)		
			As-Found [2]	Performance [2]	As-Left [2]
Pre-Op	8/23/79	Mass Point	N/A	N/A	0.1137
1st Periodic	4/18/83	Mass Point	0.1446	0.1441	0.1446
2nd Periodic	8/17/86	Mass Point	0.1566	0.1527	0.1533
3rd Periodic	5/2/90	Mass Point	0.1965	0.1953	0.1965
4th Periodic	5/30/93	Mass Point	0.1482	0.1481	0.1482

Notes: (1)

All test results reported at the 95% upper confidence limit and include the leakage penalty total for all Type B and C penetrations not challenged during performance of the Type A test.

(2)

As-left acceptance criteria (<0.75 x L_a): <0.225 weight-percent / day
Performance acceptance criteria (< 1.0 x L_a) : < 0.300 weight-percent / day
As-found acceptance criteria (< 1.0 x L_a) : < 0.300 weight-percent / day

3.2 Containment Inspections

The McGuire Unit 1 steel containment vessel is examined in accordance with the requirements of an ASME Section XI Program, the Nuclear Generation Department Coating Program, and the Containment Leakage Rate Test Program, as described below.

The ASME Section XI Program requires that the steel containment vessel be examined in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE, and associated modifications and limitations imposed by 10CFR50.55a(b)(2). These examinations have been performed using the 1992 Edition with the 1992 Addenda (Reference 6) during the first inservice inspection interval, and the 1998 Edition with the 2000 Addenda (Reference 7) during the second inservice inspection interval (which commenced July 15, 2005). Details of the containment inservice inspection program are described in the McGuire Nuclear Station Second Interval Containment Inservice Inspection Plan.

The Nuclear Generation Department Coating Program requires a visual examination to be performed to assess and document the condition of Nuclear Safety Related protective coatings located inside primary containment during each refueling outage. The interior surfaces of the containment vessel are included in the scope of this examination.

The Containment Leakage Rate Test Program controls leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and as modified by approved exemptions. This program requires that accessible interior and exterior surfaces of the primary containment be visually examined for structural deterioration which may affect the containment leak-tight integrity 3 times every 10 years, including during each shutdown for a Type A test, prior to initiating the Type A test. These visual examination requirements are satisfied by performing the ASME Code, Section XI, Subsection IWE, IWE-2500, Table IWE-2500-1, Examination Category E-A, Item E1.11 examinations in accordance with the McGuire Unit 1 Containment Structural Integrity Inspection procedure.

In addition to the programs identified above, McGuire Technical Specification 3.6.16 requires that the structural integrity of the Reactor Shield Building be verified by performing a visual inspection of the exposed interior and exterior surfaces 3 times every ten years, coinciding with visual examination of the steel containment vessel. Because the Reactor Shield Building acts as a secondary containment and protects the primary containment from the effects of weather, these visual examinations help to assure the integrity of the primary containment.

Examination results from the above programs have revealed no significant degradation that could affect either the containment vessel structural integrity or leak-tightness. Below is a summary of containment examinations completed on Unit 1 since March 12, 2003 when Duke's License Amendment Request extending the ILRT frequency from 10 to 15 years was approved.

<u>Refueling Outage 1EOC16 (2004) Examinations</u>

1. A general visual examination was performed on Accessible Surface Areas in accordance with the ASME Code, Section XI, Subsection IWE (1992 Edition with the 1992 Addenda), IWE-2500, Table IWE-2500-1, Examination Category E-A, Item E1.11. The results of this examination revealed no unacceptable conditions. The following is a summary of the more significant results from this examination.

    a. Moisture barrier (sealant) was observed to be missing or in need of corrective maintenance at a number of locations around the base of the containment vessel on the exterior side of the vessel at the embedment zone interface. No evidence of moisture intrusion was noted at these locations, and these conditions were noted in the corrective action program.

    b. Boric acid crystals from prior leakage from several components were observed on containment vessel surfaces near the base of the containment vessel on the exterior side. These conditions were noted in the corrective action program, and the affected areas were cleaned and reinspected with no significant degradation observed on containment vessel surfaces.

    c. Various locations were identified where protective coatings maintenance was recommended on the exterior side of the containment vessel. Areas that warranted corrective coatings maintenance were noted in the corrective action program. These conditions are not indicative of unacceptable degradation and were documented to ensure routine coatings maintenance was scheduled and performed

2. VT-3 visual examinations were performed on selected Moisture Barriers in accordance with the ASME Code, Section XI, Subsection IWE (1992 Edition with the 1992 Addenda), IWE-2500, Table IWE-2500-1, Examination Category E-D, Item E5.30. The results of these examinations revealed no unacceptable conditions.
3. A general visual examination was performed on accessible interior and exterior surfaces of the Reactor Shield Building. The examination results were acceptable with no significant degradation observed.
4. The Coating Program visual examinations revealed areas where corrective coatings maintenance was warranted, but no significant degradation was observed on the containment vessel surfaces.

Refueling Outage 1EOC17 (2005) Examinations

1. VT-1 visual examinations were performed on a percentage of containment pressure retaining bolted connections, as required by Duke Relief Request Serial #03-GO-010 (Reference 8). Approximately 35% of all bolted connections were examined with no unacceptable conditions observed.
2. VT-3 visual examinations were performed on the Containment Lower Air Lock Barrel Supports in accordance with the ASME Code, Section XI, Subsection IWF (1998 Edition with the 2000 Addenda), IWF-2500, Table IWF-2500-1, Examination Category F-A, Item F1.40. The examination results were acceptable.
3. The Coating Program visual examinations revealed areas where corrective coatings maintenance was warranted, but no significant degradation was observed on the containment vessel surfaces.

The programs described above will continue to provide reasonable assurance that the primary containment structural and leak-tight integrity will be maintained until the overall leak-tightness of the containment is verified during the next scheduled Type A test. Although a Type A Test will not be conducted during Refueling Outage 1EOC18, a general visual examination of containment accessible surface areas is scheduled to be performed in accordance with the ASME Code, Section XI, Subsection IWE (1998 Edition with the 2000 Addenda) IWE-2500, Table IWE-2500-1, Examination Category E-A, Item E1.11

In addition to the discussion above on Containment Inspections it should also be noted, as described in the September 25, 2002 letter from Duke to the NRC (Reference 9), the testing frequency for penetrations using seals and gaskets to assure containment leak tight integrity is not affected by this requested extension to the Type A test interval from 15 years to approximately 15.5 years. In addition, although McGuire is only required to test the bellows following an ILRT, a more conservative approach (test plan) has been implemented. McGuire has developed and is presently utilizing a supplementary testing program that tests one-third of the bellows each outage. Under this program all bellows have now been tested.

3.3 Risk Discussion

The proposed amendments are submitted on a risk-informed basis as described in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002 (Reference 4).

Duke has completed risk assessments of the proposed amendments for McGuire. These assessments use the guidance provided in EPRI TR-104285 (Reference 10) and the process identified in NUREG-1493 (Reference 11) to evaluate the risk impact of the ILRT extension requests. Additionally, the assessments compare the results to guidance contained in Regulatory Guide 1.174. The assessment considers three risk metrics – Person-Rem risk, Large Early Release Frequency (LERF), and Conditional Containment Failure Probability (CCFP). There is no impact on Core Damage Frequency. Based on the results of the assessments, the extension request has a small but acceptable increase in risk. It should also be noted that a delay in 1EOC19 will not substantially increase the risk associated with delaying the ILRT as demonstrated by the 1 in 18yr and 1 in 20yr data provided in tables 3a to 3d.

The assessment uses the results of the McGuire Revision 2 (Reference 12) and Revision 3a (Reference 13) Probabilistic Risk Assessments (PRAs). The McGuire Revision 2 PRA is a full scope, Level 3 PRA. The McGuire Revision 3a PRA is a Level 1 PRA with a separate LERF model. Therefore, the McGuire Revision 2 PRA was used to calculate the Release Categories, Person-Rem Estimates, LERF, and the CCFP. The McGuire Revision 3a PRA was used to recalculate LERF estimates.

The Person-Rem estimates and CCFP were calculated using internal events only. The LERF estimates were calculated using both internal and external events.

The containment end-states developed in the PRA were assigned to each of the EPRI Accident Classes identified in EPRI TR-104285 (Reference 10). This information is contained in Table 2 below.

Table 2
McGuire PRA Revision 2 Risk Results Summary

Accident Class	Frequency (yr^{-1})	Person-Rem[a]	Person-Rem (yr^{-1})	Comments
1	1.72E-05	1.97E+03	3.38E-02	
2	5.33E-08	2.14E+05	1.14E-02	
3				Not Developed in the MNS PRA, however, it is developed in the text below.
4				Not Analyzed
5				Not Analyzed
6				Included in Accident Class 2
7	1.06E-05	3.11E+05	3.30E+00	
8	2.46E-07	1.11E+07	2.73E+00	
Total	2.81E-05		6.08E+00	

a Frequency weighted person-rem = (Sum of End-State Person-Rem Risk)/(Sum of End-State Frequencies)

Accident Class 3 is the EPRI Accident Class that contains leakage and/or containment failure where the response is affected by ILRT. For this study, Class 3 was divided into two groups. Class 3a represents a small leak that is less than 10 x L_a, (or 3 weight-percent/day). Class 3b represents a much larger leak that contributes to LERF. The probability of leakage associated with Class 3 is assumed to be proportional to the time between tests. The probability of Class 3a was estimated using data from NUREG-1493 (Reference 11). NUREG-1493 found that there have been five failed ILRTs out of 180 ILRTs that could only have been detected by an ILRT. Based on this data, the Class 3a probability is approximately 0.03.

The Class 3b probability was estimated using the Jeffrey's "non-informative prior distribution" (Reference 14).

$$\text{Failure Probability} = \frac{\text{Number of Failures (0)} + 1/2}{\text{Number of Tests (180)} + 1}$$

The data for Class 3b consists of zero failures out of 180 ILRTs. The resulting probability is approximately 0.003. These values were used to estimate the frequencies of Class 3a and Class 3b.

For each Accident Class, the population dose and LERF were estimated. For Class 3a, the population dose is assumed to be 10 times the McGuire PRA no containment failure dose (the no containment failure end-states assume that containment leaks at 1 x L_a). For Class 3b, the population dose was assumed to be the same as the population dose for the isolation failure end-states.

The Accident Classes in Table 2 can be placed into two groups – those that are LERF and those that are not LERF. Since Class 3b represents LERF, an estimate of LERF can be made by multiplying the probability of Class 3b by the frequency of accident classes that are not LERF. The off-site consequences associated with Class 3a are assumed to be small and do not impact LERF.

The CCFP was calculated using the following equation:

$$CCFP = 1 - \frac{\text{Intact Containment Frequency}}{\text{Total Core Damage Frequency}}$$

The risk metrics were calculated for each of the following test intervals:

- 3 tests in 10 years – original requirements for ILRT
- 1 test in 10 years – current test interval
- 1 test in 15 years – current one-time interval extension
- 1 test in 15.5 years – proposed one-time interval extension
- 1 test in 18 years – sensitivity case
- 1 test in 20 years – sensitivity case

The results of the McGuire ILRT risk assessment is contained in Tables 3a-d.

Table 3a
Summary of Person-Rem Risk Results

Case	Person-Rem Risk (yr^{-1})			
	Total	**Increase Relative to Baseline**	**Increase Relative to 1 in 10 Years**	**Increase Relative to 1 in 15 Years**
3 per 10 Yr (baseline)	6.10E+00			
1 per 10 yr	6.13E+00	2.92E-02		
1 per 15 yr	6.15E+00	5.12E-02	2.19E-02	
1 per 15.5 yr	**6.15E+00**	**5.33E-02**	**2.40E-02**	**2.09E-03**
1 per 18 yr	6.16E+00	6.42E-02	3.50E-02	1.31E-02
1 per 20 yr	6.17E+00	7.31E-02	4.39E-02	2.19E-02

Table 3b
Summary of LERF Results Using the MNS Rev. 2 PRA

Case	LERF			
	Total	Increase Relative to Baseline	Increase Relative to 1 in 10 Years	Increase Relative to 1 in 15 Years
3 per 10 Yr (baseline)	4.13E-06			
1 per 10 yr	4.38E-06	2.51E-07		
1 per 15 yr	4.57E-06	4.39E-07	1.88E-07	
1 per 15.5 yr	**4.59E-06**	**4.57E-07**	**2.06E-07**	**1.79E-08**
1 per 18 yr	4.68E-06	5.51E-07	3.00E-07	1.12E-07
1 per 20 yr	4.76E-06	6.27E-07	3.76E-07	1.88E-07

Table 3c
Summary of LERF Results Using the MNS Rev. 3a PRA

Case	LERF			
	Total	Increase Relative to Baseline	Increase Relative to 1 in 10 Years	Increase Relative to 1 in 15 Years
3 per 10 Yr (baseline)	3.64E-06			
1 per 10 yr	3.83E-06	1.91E-07		
1 per 15 yr	3.98E-06	3.33E-07	1.43E-07	
1 per 15.5 yr	**3.99E-06**	**3.47E-07**	**1.57E-07**	**1.36E-08**
1 per 18 yr	4.06E-06	4.18E-07	2.28E-07	8.51E-08
1 per 20 yr	4.12E-06	4.76E-07	2.86E-07	1.43E-07

Table 3d
Summary of CCFP Results

Case	CCFP			
	Total	Increase Relative to Baseline	Increase Relative to 1 in 10 Years	Increase Relative to 1 in 15 Years
3 per 10 Yr (baseline)	39.05%			
1 per 10 yr	39.40%	0.34%		
1 per 15 yr	39.65%	0.60%	0.26%	
1 per 15.5 yr	**39.68%**	**0.62%**	**0.28%**	**0.02%**
1 per 18 yr	39.81%	0.75%	0.41%	0.15%
1 per 20 yr	39.91%	0.86%	0.51%	0.26%

3.3.1 Person-Rem Analysis

The first risk measure that is considered in this analysis is person-rem risk. The increase in person-rem risk for extending the Type A test frequency from 1 in 10 years ranges from 2.4E-02 person-rem/yr (0.39%) for 1 in 15.5 years to 4.4E-02 person-rem/yr (0.71%) for 1 in 20 years. Extending the Type A test frequency for the different cases does not have a significant impact on person-rem risk.

The person-rem risk results in this analysis are slightly higher than the results in NUREG 1493 (Reference 11) and EPRI TR-104285 (Reference 10). These two previous assessments found that extending the Type A test interval results in a person-rem risk increase that is much less than 1% (0.02% to 0.14%). The main difference in the person-rem risk increase calculated in this analysis and the previous analysis is the assumption of the dose associated with Class 3b. Neither the NUREG study nor the EPRI study considers a very large leak that is sufficient to result in LERF. These studies assumed that a Type A failure would result in a leak rate of approximately $2L_a$. However, since Class 3b is supposed to represent LERF, then the person-rem associated with Class 3b is very large compared to the person-rem for a $2L_a$ leak. The leak rate and the dose associated with Class 3b are more representative of a hole in containment versus a leak in containment.

3.3.2 LERF Analysis

The second risk measure considered in this analysis is Large Early Release Frequency (LERF). A comparison of the population dose for Type A failures to other LERF accident classes indicates that the estimated leak rate based on historical data is well below the leak rate for LERF. The LERF Analysis was performed using both the McGuire Revision 2 and Revision 3a PRA results.

a. *McGuire Revision 2 LERF Analysis*

The estimated increase in LERF using the McGuire Revision 2 PRA results due to McGuire extending the ILRT interval of 1 in 10 years ranges from 2.06E-07/yr for 1 test in 15.5 years to 3.76E-07/yr for 1 test in 20 years. Changes in LERF that are less than 1E-07 per year are considered very small in Reg. Guide 1.174 (Reference 4). These values are above the Reg. Guide 1.174 value for a very small change. When the increase in LERF is between 1.0E-07/yr and 1.0E-06/yr, the total LERF must be considered. For these situations, the total LERF must be less than 1.0E-05/yr. The total LERF ranges from 4.59E-06/yr to 4.76E-06/yr (1 in 15.5 years to 1 in 20 years).

b. *McGuire Revision 3a LERF Analysis*

The MNS Rev. 3a PRA CDF calculation includes both internal and external events. The MNS Rev. 3a PRA LERF calculation includes all events except for Seismic events. In order to calculate the seismic contribution to LERF, the Seismic split fraction from the MNS Rev. 2 PRA was applied to the Rev. 3a Seismic CDF. This method of estimating the Seismic contribution to LERF is acceptable because the overall Seismic CDF decreased in Rev. 3a of the PRA and also because the increase in LERF and the total LERF are not overly sensitive to changes in the split fraction.

The estimated increase in LERF using the McGuire Revision 3a PRA results due to McGuire extending the ILRT interval of 1 in 10 years ranges from 1.57E-07/yr for 1 test in 15.5 years to 2.86E-07/yr for 1 test in 20 years. Changes in LERF that are less than 1E-07 per year are considered very small in Reg. Guide 1.174 (Reference 4). These values are above the Reg. Guide 1.174 value for a very small change. When the increase in LERF is between 1.0E-07/yr and 1.0E-06/yr, the total LERF must be considered. For these situations, the total LERF must be less than 1.0E-05/yr. The total LERF ranges from 3.99E-06/yr to 4.12E-06/yr (1 in 15.5 years to 1 in 20 years).

Therefore, even with the conservative estimates of LERF, extending the ILRT at McGuire from 1 test in 10 years to 1 test in 15.5 years for both McGuire Revision 2 and Revision 3a results is acceptable by the Reg. Guide 1.174 guidelines.

In this analysis, the contribution of Type A leakage events to LERF is not negligible. One reason for this result is that the Class 3b frequency is based on limited data. The Class 3b frequency is based on zero failures out of 180 ILRTs. Collection of more data concerning containment leaks would provide a better estimate of the Class 3b frequency. In addition, the assumption that Class 3b represents LERF is conservative. The contribution to LERF due to leakage events is most likely smaller than the contribution estimated in this analysis.

Furthermore, other analysis and submittals have suggested a lower probability for a Class 3b failure. The Seabrook submittal (Reference 15) used a similar method to calculate the Class 3b probability. For the LERF portion of Class 3b, the Seabrook analysis reduced the probability by a factor of 40. In 2003 EPRI conducted a study on the risk impact contribution for revised containment leak rate testing intervals (Reference 16). The EPRI report suggests a LERF probability that is at least a factor of 10 lower than what is calculated in this analysis. The analysis also does not account for visual inspections and other detection means. These activities increase the likelihood that a containment leak would be detected prior to an event; therefore the risk associated with a leak going undetected between ILRT is somewhat lower than calculated. Considering these items the LERF is likely to be less than the calculated values.

3.3.3 CCFP Analysis

The analysis calculates the CCFP for the Type A test intervals. The change in CCFP from 1 in 10 years ranges from 0.28% for 1 in 15.5 years to 0.51% for 1 in 20 years. The extension cases have very little impact on CCFP.

Based on the results of these three analyses (Person-Rem, LERF, and CCFP), extending the McGuire ILRT frequency result has an acceptable impact on plant risk.

3.3.4 PRA Quality Statement - McGuire

Duke periodically evaluates changes to the plant with respect to the assumptions and modeling in the McGuire PRA. The original McGuire PRA was initiated in March 1982 by Duke Power Company staff with Technology for Energy Corporation as a contractor. Law Engineering Testing Company and Structural Mechanics Associates provided specific input to the seismic analysis. It was a full scope Level 3 PRA with internal and external events. A peer review of the draft PRA was conducted by Electric Power Research Institute's Nuclear Safety Analysis Center (NSAC) in May 1983 (Reference 17). The final study, which incorporated the comments of the peer review, was completed in July 1984 and resulted in an internal Duke report (Reference 18) as Revision 0 to the PRA. In January 1988, Duke Power Company initiated a complete review and update of the original study.

On November 23, 1988, the NRC issued Generic Letter 88-20 (Reference 19), which requested that licensees conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plants. The McGuire response to GL 88-20 was provided by letter dated November 4, 1991 (Reference 20). McGuire's response included an updated McGuire PRA (Revision 1) study which was the culmination of the review and update which began in January 1988.

The McGuire PRA Revision 1 study and the IPE process resulted in a comprehensive, systematic examination of McGuire with regard to potential severe accidents. The McGuire study was again a full-scope, Level 3 PRA with analysis of both the internal and external events. This examination identified the most likely severe accident sequences, both internally and externally induced, with quantitative perspectives on likelihood and fission product release potential. The results of the study prompted changes in equipment, plant configuration and enhancements in plant procedures to reduce vulnerability of the plant to some accident sequences of concern.

As part of the Generic Letter 88-20 IPE process, the NRC conducted an audit of the human reliability analysis of the McGuire IPE during the period July 28 – 30, 1993. By letter dated June 30, 1994 (Reference 21), the NRC provided a Staff Evaluation of the internal events portion of the above McGuire IPE submittal which included the results of the human reliability analysis audit. The conclusion of the NRC letter [page 15] states:

"The staff finds the licensee's IPE submittal for internal events including internal flooding essentially complete, with the level of detail consistent with the information requested in NUREG-1335. Based on the review of the submittal, and audit of "tier 2" supporting information, the staff finds reasonable the licensee's IPE conclusion that no severe accident vulnerabilities exist at McGuire."

In response to Generic Letter 88-20, Supplement 4, Duke completed an Individual Plant Examination of External Events (IPEEE) for severe accidents. This IPEEE was submitted to the NRC by letter dated June 1, 1994 (Reference 22). The report contained a summary of the methods, results and conclusions of the McGuire IPEEE program. The IPEEE process and supporting McGuire PRA included a comprehensive, systematic examination of severe accident potential resulting from external initiating events. By letter dated February 16, 1999, (Reference 23) the NRC provided an evaluation of the IPEEE submittal. The conclusion of the NRC letter [page 6] states:

"On the basis of the overall review findings, the staff concludes that: (1) the licensee's IPEEE is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407), and (2) the IPEEE results are reasonable given the MNS design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the MNS IPEEE has met the intent of Supplement 4 to GL 88-20 and the resolution of specific generic safety issues discussed in the SER."

In 1997, McGuire initiated Revision 2 of the 1991 IPE and provided the results to the NRC in 1998 (Reference 24). Revision 3 of the McGuire PRA was completed in July 2002 and Revision 3a was completed in February 2005. Revision 3 was a comprehensive revision to the PRA models and associated documentation. The objectives of this update were as follows:

- To ensure the models comprising the PRA accurately reflect the current plant, including its physical configurations, operating procedures, maintenance practices, etc.
- To review recent operating experience with respect to updating the frequency of plant transients, failure rates, and maintenance unavailability data.

- To correct items identified as errors and implement PRA enhancements as needed.
- To address areas for improvement identified in the recent McGuire PRA Peer Review.
- To utilize updated Common Cause Analysis data and Human Reliability Analysis data.

PRA maintenance encompasses the identification and evaluation of new information into the PRA and typically involves minor modifications to the plant model. PRA maintenance and updates as well as guidance for developing PRA data and evaluation of plant modifications, are governed by Workplace Procedures.

Approved workplace procedures address the quality assurance of the PRA. One way the quality assurance of the PRA is ensured is by maintaining a set of system notebooks on each of the PRA systems. Each system PRA analyst is responsible for updating a specific system model. This update consists of a comprehensive review of the system including drawings and plant modifications made since the last update as well as implementation of any PRA change notices that may exist on the system. The analyst's primary focal point is with the system engineer at the site. The system engineer provides information for the update as needed. The analyst will review the PRA model with the system engineer and as necessary; conduct a system walk down with the system engineer.

The system notebooks contain, but are not limited to, documentation on system design, testing and maintenance practices, success criteria, assumptions, descriptions of the reliability data, as well as the results of the quantification. The system notebooks are reviewed and signed off by a second independent person and are approved by the manager of the group.

When any change to the PRA is identified, the same three-signature process of identification, review, and approval is utilized to ensure that the change is valid and that it receives the proper priority.

In January 2001, an enhanced manual configuration control process was implemented to more effectively track, evaluate, and implement PRA changes to better ensure the PRA reflects the as-built, as-operated plant. This process was further enhanced in July 2002 with the implementation of an electronic PRA change tracking tool.

3.3.5 Peer Review Process- McGuire

Between October 23-27, 2000, McGuire participated in the Westinghouse Owners Group (WOG) PRA Certification Program. This review followed a process that was originally developed and used by the Boiling Water Reactor Owners Group (BWROG) and subsequently broadened to be an industry-applicable process through the Nuclear Energy Institute (NEI) Risk Applications Task Force. The resulting industry document, NEI-00-02 (Reference 25), describes the overall PRA peer review process. The Certification/Peer Review process is also linked to the ASME PRA Standard (Reference 26).

The objective of the PRA Peer Review process is to provide a method for establishing the technical quality and adequacy of a PRA for a range of potential risk-informed plant applications for which the PRA may be used. The PRA Peer Review process employs a team of PRA and system analysts, who possess significant expertise in PRA development and PRA applications. The team uses checklists to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA being reviewed. One of the key parts of the review is an assessment of the maintenance and update process to ensure the PRA reflects the as-built plant.

The review team for the McGuire PRA Peer Review consisted of six members. Three of the members were PRA personnel from other utilities. The remaining three were industry consultants. Reviewer independence was maintained by assuring that none of the six individuals had any involvement in the development of the McGuire PRA or IPE.

A summary of some of the McGuire PRA strengths and recommended areas for improvement from the peer review are as follows:

Strengths

- Good Summary Report write-up with insights
- Good system notebooks
- Rigorous Level 2 & 3 PRA Model
- Integrated internal and external events model
- Up-to-date plant database using Maintenance Rule
- Ongoing PRA staff interaction with plant staff, plant staff reviews
- PRA personnel knowledge of plant good

Recommended Areas for Improvement

- Better integration of sequences and recoveries within quantification process needed
- Need to review treatment of events requiring time-phasing in the modeling
- Better approach to closing the loop on PRA update items (tracking of errors/mods) needed
- More thorough, systematic approach to HRA screening values and common cause modeling needed
- Need to update the PRA model to be more in line with current practices and expectations for state-of-the-art PRA

The significance levels of the WOG Peer Review Certification process have the following definitions:

A. Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process.

B. Important and necessary to address but may be deferred until the next PRA update.

Based on the PRA peer review report, the McGuire PRA received six Fact and Observations (F&O) with the significance level of “A” and 31 F&O with the significance level of “B.” All six of the “A” F&O have been resolved and changes have been incorporated into McGuire PRA Revision 3a, the current PRA model. The “B” F&O have been reviewed and prioritized for incorporation into the PRA. Twelve of the “B” F&O have already been incorporated into Revision 3a of the PRA.

It is expected that the remaining F&O will be resolved and incorporated into Revision 4 of the PRA. The remaining open “B” F&O were reviewed with respect to any impact on the proposed LAR submittal. Thirteen of the open “B” F&O were identified as having no impact on the proposed TS changes. A discussion of the remaining peer review items related to this TS change and their resolution is provided in Table 4. It was determined that these have no significant impact on the proposed TS changes.

Table 4
Summary of Open McGuire B F&Os

F&O	PRA Change Form	Summary of Peer Review Fact and Observation	Basis for no Significant Impact
HR-1	M-02-0065	The pre-initiator human interactions (HIs) are not included for modeling instrument miscalibration events. Further, no systematic process to identify pre-initiator human actions is identified in the Human Reliability Analysis (HRA). If a systematic process was followed to identify pre-initiator actions in the HRA, document the process followed. If such a process was not followed, develop a process and	In general, conservative screening values were retained for the pre-initiator Human Error Probability (HEPs) since they do not contribute appreciably to the CDF or LERF. Miscalibration errors are considered to be included within these events. The impact on the LAR submittal is currently bounding and is considered not significant.

F&O	PRA Change Form	Summary of Peer Review Fact and Observation	Basis for no Significant Impact
		determine whether additional actions should be included. Provide the basis for excluding miscalibration events, or develop appropriate events for inclusion in the next update of the Probabilistic Safety Assessment (PSA) model.	
HR-3	M-02-0067	Some of the dynamic human events are evaluated using the Human Cognitive Reliability (HCR) model. For these, the only performance shaping factors (PSFs) considered are time available and operator response time. The HRA documentation lists the potential effects of additional PSFs, such as operator experience, but they do not appear to have been applied in the quantification of human interaction (HI) events. Consider the effects of the various PSFs in the HCR model.	This item is a methodology issue which is not expected to significantly impact overall risk results. Operator response times implicitly include performance shaping factors, since they are based on simulator runs or expert elicitation. No specific problems regarding the HRA analysis itself were identified by the peer review team. Therefore, the impact on the LAR submittal is not significant.
ST-1	M-02-0023	The current Interfacing-System Loss of Coolant Accident (ISLOCA) analysis may overstate the ISLOCA CDF contribution. Consider implementing a more recent methodology, including the dynamic effects of valve rupture on piping integrity and possibly incorporating the results of the ongoing risk-informed inservice inspection of piping study if appropriate, to ensure that the McGuire approach is sufficiently realistic.	This item concerns the potential that the current ISLOCA model may overstate the ISLOCA CDF contribution. If so, then a more detailed analysis would yield a smaller calculated total LERF value. Therefore, the impact on the LAR submittal is currently bounding and is considered not significant.

F&O	PRA Change Form	Summary of Peer Review Fact and Observation	Basis for no Significant Impact
TH-1	M-02-0034	Success criteria for some systems are supported by Modular Accident Analysis Program (MAAP) runs with MAAP 3b, Version 16. This version of MAAP has been found to have deficiencies which can impact conclusions and results. Re-run selected analyses with a later version of the MAAP code or make use of other transient analysis results.	McGuire has upgraded to using MAAP Version 4.0.5. The McGuire success criteria are generally consistent with other similar plants. Also, the reconstitution of the success criteria database is expected to confirm the vast majority of the success criteria. Therefore few changes are anticipated and no significant change in the CDF or LERF is expected.
TH-3	M-02-0048	Success criteria analyses were not done for the range of possible plant conditions to which they are applied. For example, medium LOCA success criteria analyses are done for a 3.5 inch break, although the medium LOCA is defined as a 2 to 5 inch break. The combinations of systems and operator recoveries that are defined as success at 3.5 inches may not be success at 2 inches or at 5 inches. This issue also applies to large and small LOCAs. Also, MAAP is not an appropriate code to use in performing analyses for rapid blowdown events such as large and some medium LOCAs. Perform success criteria analyses for a range of possible conditions for each application. Also, a code other than MAAP should be used if large and medium LOCA success criteria	McGuire has upgraded to using MAAP Version 4.0.5. The McGuire success criteria are generally consistent with other similar plants. Also, the reconstitution of the success criteria database is expected to confirm the vast majority of the success criteria. Therefore few changes are anticipated and no significant change in the CDF or LERF is expected.

F&O	PRA Change Form	Summary of Peer Review Fact and Observation	Basis for no Significant Impact
		are being defined.	
TH-4	M-02-0049	Success criteria do not appear to have been sufficiently reviewed. [The reviewers identified three apparent errors in the MAAP analyses.] Perform independent review of success criteria analyses. Verify that any identified analysis errors do not change the success criteria bases.	McGuire has upgraded to using MAAP Version 4.0.5. The McGuire success criteria are generally consistent with other similar plants. The reconstitution of the success criteria database is expected to confirm the vast majority of the success criteria. The identified apparent errors are trivial in nature and the LERF cutset was reviewed to verify that the identified systems did not affect the top cutsets. Therefore few changes are anticipated and no significant change in the CDF or LERF is expected.

3.3.6 PRA Model - McGuire

The McGuire PRA includes both internal and external events. The model includes the necessary initiating events (e.g., LOCAs, transients) to evaluate the frequency of accidents. The previous reviews of the McGuire PRA, NRC and peer reviews have not identified deficiencies related to the scope of initiating events considered.

The McGuire PRA includes models for those systems needed to estimate core damage frequency. These include all of the major support systems (e.g., ac power, service water, component cooling, and instrument air) as well as the mitigating systems (e.g., emergency core cooling). These systems are modeled down to the component level, pumps, valves, and heat exchangers. This level of detail is sufficient for this application.

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

Title 10 of the *Code* of *Federal Regulations,* Part *50,* Appendix J, was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." The use of Option B for the Type A (integrated) leakage rate testing was approved on March 21, 1997, for McGuire Unit 1 by License Amendment No. 173 (Reference 27). The use of Option B for Type B and C (local) leakage rate testing was approved on September 4, 2002, for McGuire Unit 1 by License Amendment No. 207 (Reference 28). These amendments modified TS Section 5.5.2, to allow Type A, B, and C testing to be performed in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995 (Reference 1). RG 1.163 specifies a method acceptable to NRC for complying with Option B and approves, with certain exceptions, the use of Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 2), and American National Standards Institute / American Nuclear Society (ANSI / ANS) standard 56.8-1994 (Reference 29).

The overall integrity of the containment structure is verified by a Type A ILRT and the integrity of the penetrations is verified by Type B and Type C local leak rate tests as required by 10 CFR Part 50, Appendix J. These tests are performed to verify the essentially leak-tight characteristics of the containment structure at the design basis accident pressure. Based on the last two Type A ILRTs for McGuire unit 1 and the risk assessment results presented in the September 25, 2002 Duke submittal (Reference 9), in accordance with 10 CFR Part 50, Appendix J, Option B, the current ILRT interval is 15 years.

The adoption of the Option B requirements did not alter the basic method by which leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in ILRTs. Frequency is based upon an evaluation which examines the '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 changes contained in this amendment request will not directly result in an increase in containment leakage. The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995. 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.

EPRI Research Project Report TR-104285 (Reference 10) documents a similar study. In addition, as stated in the Safety Evaluation by the Office of Nuclear Reactor Regulation related to amendment no 211 for McGuire Nuclear Station Unit 1:

> "The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for McGuire early in the plant's life, required an ILRT test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak that was detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of the leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the PWR and boiling water reactor representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in increased public dose."

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 (Reference 2) 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 the maximum allowable leakage rate (1.0 x La) and consideration of the performance factors in NEI 94-01, Section 11.3). Based on the last two ILRTs for McGuire unit 1, the current interval for each unit is once every fifteen years.

Previous McGuire Type A test results have shown leakage to be below the leakage limits. Refer to Table 1 for a summary of Type A test results for McGuire. Accordingly, the proposed extension of the Type A test for McGuire represents minimal risk for increased leakage. The risk is further minimized by continued Type B and Type C testing. Also, the McGuire In-service Inspection (ISI) programs and maintenance rule inspections provide additional confidence in containment structural integrity and leak tightness.

4.2 PRECEDENTS

On December 23, 2005 the NRC issued Amendment No. 140 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No.2, to extend the date for the next Appendix J, Type A test for approximately 6-months, until the end of the SL2-17 refueling outage (Reference 30). In addition, on March 24, 2006 the NRC issued Amendment No. 108 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1 to permit a one-time, 6 month, addition to the currently approved 5-year extension to the 10-year test interval for the containment integrated leak rate test (Reference 31).

4.3 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The following discussion is a summary of the evaluation of the changes contained in the proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or

3. Involve a significant reduction in a margin of safety.

4.3.1 First Standard

The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed extension to the Type A testing intervals cannot increase the probability of an accident previously evaluated since extension of the intervals is not a physical plant modification that could alter the probability of accident occurrence, nor is it an activity or modification by itself that could lead to equipment failure or accident initiation. The proposed extension to the Type A testing intervals does not result in a significant increase in the consequences of an accident as documented in NUREG-1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and Type C tests. It concludes that reducing the Type A testing frequency to once per twenty years leads to an imperceptible increase in risk.

McGuire provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. Prior Type A tests for McGuire Unit 1 identified containment leakage within acceptance criteria, indicating a very leak tight containment. Inspections required by the ASME Code are also performed in order to identify indications of containment degradation that could affect leak tightness. Separately, Type B and Type C testing, required by TS, identify any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the Type A test intervals will not represent a significant increase in the consequences of an accident.

4.3.2 Second Standard

The proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revisions to the McGuire TS add a one-time extension to the current interval for Type A testing. The current test interval of fifteen years, based on past performance, would be extended on a one-time basis to approximately fifteen and a half years from the last Type A test. The proposed extension to the Type A test interval does not create the possibility of a new or different type of accident since there are no physical changes being made to the plants and there are no changes to the operation of the plants that could introduce a new failure mode.

4.3.3 Third Standard

The proposed amendment will not involve a significant reduction in a margin of safety. The proposed revisions to the McGuire TS add a one-time extension to the current interval for Type A testing. The current test interval of fifteen years, based on past performance, would be extended on a one-time basis to approximately fifteen and a half years from the last Type A test. The proposed extension to Type A test intervals will not significantly reduce the margin of safety. The NUREG-1493 generic study of the effects of extending containment leakage testing intervals found that a twenty-year interval resulted in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leakage rate contributes about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal effect on this risk, since 95 percent of the Type A detectable leakage paths would already be detected by Type B and Type C testing. Similar proposed changes have been previously reviewed and approved by the NRC, and they are applicable to McGuire.

Based upon the preceding discussion, Duke Energy Corporation has concluded that the proposed amendments do not involve a significant hazards consideration.

4.4 CONCLUSION

Extending the ILRT frequency would result in a small increase in risk. The increase in LERF from the proposed extension is acceptable under the criteria in RG 1.174. An ILRT extension is not likely to affect the detection of degradation of the containment.

In conclusion, based on the considerations discussed previously, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.22(b), an evaluation of these license amendment requests has been performed to determine whether or not they meet the criteria for categorical exclusion set forth in 10 CFR 51.22(c) (9) of the regulations.

This amendment to the McGuire TS allows for a one-time extension of ILRT intervals from fifteen to approximately fifteen and one half years from the date of the last successful test. Implementation of this amendment will have no adverse impact upon McGuire unit 1; neither will it contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure. It has been determined there is:

1. No significant hazards consideration,
2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and
3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, the amendment to the McGuire TS meets the criteria of 10 CFR 51.22(c) (9) for categorical exclusion from an environmental impact statement.

6.0 REFERENCES

1. Regulatory Guide (RG) 1.163 "Performance-Based Containment Leak-Test Program" September 1995
2. NEI 94-01 Rev 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,"
3. March 12, 2003 Letter from the NRC (Robert E. Martin) to Duke Power Company (Dhiaa Jamil), issuing Amendment 211 facility operating license MPF-9 and amendment 192 to facility operating license MPF-17 to allow a one-time 5-year change in the Appendix J, Type A containment integrated leakage rate test interval
4. Regulatory Guide (RG) 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-specific Changes to the Licensing Basis"
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