September 28, 2004

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - AMENDMENT RE: ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MC0247)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 279 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated July 28, 2003, as supplemented on May 20, 2004.

The amendment revises TS 5.5.6,"Primary Containment Leakage Rate Testing Program," to allow a one-time extension of the interval between the Type A, integrated leakage rate tests, from 10 years to no more than 15 years; that is no later than March 7, 2010.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures: 1. Amendment No. 279 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

September 28, 2004

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - AMENDMENT RE: ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MC0247)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 279 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated July 28, 2003, as supplemented on May 20, 2004.

The amendment revises TS 5.5.6,"Primary Containment Leakage Rate Testing Program," to allow a one-time extension of the interval between the Type A, integrated leakage rate tests, from 10 years to no more than 15 years; that is no later than March 7, 2010.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures: 1. Amendment No. 279 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

TSs:
Package Number: ML042720232
Accession Number: ML042720209

OFFICE	PDI-1\PM	PDI-1\LA	EMEB\SC	SPSB\SC	OGC	PDI-1\SC
NAME	PMilano	SLittle	KManoly	RDennig	HMcGurren	RLaufer
DATE	8/02/04	8/02/04	8/03/04	8/04/04	08/10/04	09/28/04

Official Record Copy

DATED: September 28, 2004

AMENDMENT NO. 279 TO FACILITY OPERATING LICENSE NO. DPR-59 FITZPATRICK

PUBLIC
PDI-1 R/F
R. Laufer
R. Dennig
K. Manoly
J. Pulsipher
R. Palla
T. Cheng
OGC
G. Hill (2)
T. Boyce
ACRS
S. Little
C. Bixler, RI

cc: Plant Service list

FitzPatrick Nuclear Power Plant

cc:

Mr. Gary J. Taylor
Chief Executive Officer
Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213

Mr. John T. Herron
Sr. VP and Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Theodore A. Sullivan
Site Vice President
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. Kevin J. Mulligan
General Manager, Plant Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. Danny L. Pace
Vice President, Engineering
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Brian O'Grady
Vice President, Operations Support
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John F. McCann
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Resident Inspector's Office
James A. FitzPatrick Nuclear Power Plant
U. S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, NY 13093

Ms. Charlene D. Faison
Manager, Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Michael J. Colomb
Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. William Maquire
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. Andrew Halliday
Manager, Regulatory Compliance
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, NY 13126

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

FitzPatrick Nuclear Power Plant

cc:

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Oswego County Administrator
Mr. Steven Lyman
46 East Bridge Street
Oswego, NY 13126

Mr. Peter R. Smith, President
New York State Energy, Research,
and Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
New York State Dept. of Public Service
3 Empire State Plaza
Albany, NY 12223-1350

Mr. John M. Fulton
Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Ken L. Graesser
BWR SRC Consultant
38832 N. Ashley Drive
Lake Villa, IL 60046

Mr. Jim Sniezek
Nuclear Management Consultant
5486 Nithsdale Drive
Salisbury, MD 21801

Mr. Ron Toole
BWR SRC Consultant
1282 Valley of Lakes
Box R-10
Hazelton, PA 18202

Ms. Stacey Lousteau
Treasury Department
Entergy Services, Inc.
639 Loyola Avenue
Mail Stop L-ENT-15E
New Orleans, LA 70113

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 279
License No. DPR-59

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated July 28, 2003, as supplemented on May 20, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 279, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 28, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 279

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
5.5-5	5.5-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 279 TO FACILITY OPERATING LICENSE NO. DPR-59

ENTERGY NUCLEAR OPERATIONS, INC.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated July 28, 2003, as supplemented on May 20, 2004, Entergy Nuclear Operations, Inc. (the licensee) submitted a request for a change to the James A. FitzPatrick Nuclear Power Plant (JAF) Technical Specifications (TSs). The requested change would revise TS 5.5.6, "Primary Containment Leakage Rate Testing Program," to allow a one-time extension of the interval between the containment integrated leakage rate tests (ILRTs), from 10 years to no more than 15 years. The supplement dated May 20, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 27, 2004 (69 FR 44696).

2.0 REGULATORY EVALUATION

Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) specifies the containment leakage test requirements which provide, in part, for the periodic verification by tests of the leak-tight integrity of the primary reactor containment. Appendix J includes two options, A and B, either of which can be chosen for meeting the requirements of this appendix. 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." On October 4, 1996, the Nuclear Regulatory Commission (NRC) staff issued Amendment 234 to permit the implementation of Option B for JAF (Reference 2).

Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. JAF TS 5.5.6 requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with one exception listed in the TSs. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) Report 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

3.0 TECHNICAL EVALUATION

A Type A test is an overall or ILRT of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval at least once in 10 years, based on satisfying the performance factors in NEI 94-01, Section 11.3 and an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate is less than $1.0L_a$). In order to meet the NEI 94-01 requirement, the licensee stated that two consecutive Type A tests were previously performed in June 1990 and March 1995, which demonstrated that the leakage rates from these two tests are less than the allowable. Based on these successful Type A tests and the requirements of 10 CFR Part 50, Appendix J, Option B, the current interval requirement is 10 years for JAF. The current 10-year interval for Type A testing ends at JAF on March 7, 2005. There is also a provision for extending the test interval an additional 15 months in certain circumstances.

3.1 Proposed Change to TS 5.5.6

In its July 28, 2003, application (Reference 1), the licensee requested a change to TS 5.5.6, “Primary Containment Leakage Rate Testing Program,” that would add an exception from the guidelines of RG 1.163, regarding the Type A test interval. Specifically, the proposed TS change would state that the first Type A test performed after the March 7, 1995, Type A test (the date of the last ILRT) shall be performed no later than March 7, 2010.

The licensee also stated that the Types B and C local leakage rate tests (LLRTs), including their schedules, are not affected by this request and that the extended testing interval will not affect any American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements or Code acceptance criteria.

3.2 Inservice Inspection (ISI) of Primary Containment Integrity

JAF utilizes an 881 megawatt electric (MWe) General Electric boiling-water reactor (BWR4) with a Mark 1 type primary containment. The containment consists of two primary interconnected structures: a steel drywell, housing the reactor and related components; and a toroidal suppression chamber (torus). The drywell, which includes the major primary containment volume, is inerted with nitrogen and maintained at a nominal 1.7 psid positive pressure with respect to the torus. The primary containment is penetrated by access, piping, and electrical penetrations.

The leak rate testing requirements (ILRT and LLRTs) of Option B of Appendix J to 10 CFR Part 50, and the containment ISI requirements mandated by 10 CFR 50.55a complement each other in ensuring the leak-tightness and structural integrity of the containment. Based on its review of Type A test interval extension applications for other plants, the NRC staff had previously identified the following five general areas that the licensee was requested to address with regard to the ISI of the containment:

1. Since the submittals did not include sufficient description or summary of the containment ISI program being implemented at the plant, provide a description of the ISI methods that provide assurance that in the absence of a containment ILRT for 15 to 20 years, the containment structural and leak-tight integrity will be maintained.

2. Section IWE-1240 of the ASME Code, Section XI, requires licensees to identify the containment surface areas requiring augmented examinations. Provide the locations of the steel containment (or concrete containment liner) surfaces that have been identified as requiring augmented examination and a summary of the findings of the examinations performed.

3. For the examination of penetration seals and gaskets, and examination and testing of bolted connections associated with the primary containment pressure boundary (ASME Code Examination Categories E-D and E-G), the licensee requested relief from the requirements of the Code. As an alternative, the licensee proposed to examine the above items during the leak-rate testing of the primary containment. However, Option B of Appendix J for Type B and Type C testing (as per NEI 94-01 and RG 1.163), and the ILRT extension requested in this amendment for Type A testing, provide flexibility in the scheduling of these inspections. Provide your schedule for examination and testing of seals, gaskets, and bolted connections that provide assurance regarding the integrity of the containment pressure boundary.

4. The stainless steel bellows have been found to be susceptible to trans-granular stress corrosion cracking, and the leakage through these bellows are not readily detectable by the Type B testing (see NRC Information Notice 92-20). If applicable, provide information regarding inspection and testing of the bellows, and how their performance has been factored into the risk assessment of containment leaking to support the proposed TS change.

5. Inspections of some reinforced concrete and steel containment structures have identified degradation on uninspectable (embedded) side of the drywell steel shell and steel liner of the primary containment. These degradations cannot be found by visual examinations (i.e., VT-1 of VT-3) unless they are through the thickness of the shell or liner, or 100 percent of the uninspectable surfaces are periodically examined by ultrasonic testing. Provide information addressing how potential leakage under high pressure during core damage accidents is factored into the risk assessment related to the extension of the ILRT.

The staff's evaluation of the licensee's responses to the above areas is discussed as follows:

(1) In response to the first item, the licensee stated, in Attachment 5, "Containment Inservice Inspection Program Summary," to the July 28 application, that JAF has implemented a containment ISI program in accordance with the requirements of ASME Section XI, "Inservice Inspection," Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plant." According to the licensee, the JAF containment ISI program was developed based on the requirements of Subsection IWE of ASME Section XI, 1992 Edition including 1992 Addenda. The IWE examinations scheduled for the first period (September 28, 1987 - March 27, 2001) were performed in accordance with the requirements of the 1992 Edition, 1992 Addenda, and 10 CFR 50.55a. On the basis of the NRC staff's safety evaluation (SE) issued on May 1, 2002 (Reference 4), which approved the use of ASME Section XI, Subsection IWE, 1998 Edition (no Addenda), the licensee is going to perform the second period (March 27, 2001 - September 24, 2004) and the third period (September 28, 2004 - September 27, 2006) of ISI based on the requirements of the

1998 Edition (no Addenda) and 10 CFR 50.55a. The components subject to Subsection IWE requirements are those that make up the containment structure and its leak-tight barrier (including integral attachments), and those that contribute to its structural integrity. Specifically included are Class MC pressure retaining components, such as the drywell, torus, pressure retaining bolting, etc. The licensee also stated that there will be no change to the schedule for the ISI as a result of the extended ILRT interval. Based on its review of the information provided by the licensee, the NRC staff finds that the schedule for implementing the containment ISI program will not be affected by the requested extension of the ILRT interval (up to 15 years).

(2) For the second item related to the application of an augmented examination (required by IWE Table-2500-1, Examination Category E-C), the licensee stated, in its May 20 letter (Reference 3), that based on the ISI inspections performed, no areas of the steel containment surfaces require augmented examination and no loss of structural integrity of primary containment vessel was observed. The NRC staff finds the licensee's response reasonable and acceptable.

(3) With regard to the item related to the ISI of seals, gaskets and the pressure retaining bolting, the licensee indicated, in the May 20 letter, that in accordance with the NRC staff's SE of May 1, 2002, if a bolted connection within the IWE boundary is disassembled, a detailed visual examination will be performed once per inspection interval (10 years), consistent with the requirements of the 1992 Addenda of the ASME Code, Section XI. The detailed visual examination (VT-1) will be performed on all accessible surface areas of the bolts, studs, nuts, bushings, washers, threads in the base material, and flange ligaments between the fastener holes. In addition, a VT-1 examination will be performed if a bolted connection is disassembled at the time of a scheduled general visual examination, or at times other than a scheduled visual examination such as a maintenance activity. All accessible surface areas of the connection (bushings, threads, ligaments in the base material of flange) will be included in the examination. In order to ensure the integrity of leak tightness for seals, gaskets and bolted connections, all seals and gaskets are tested at least once every 10 years (and inspected when disassembled) in accordance with Type B, Appendix J, Option B requirements. Also, the bolting, seal and gasket inspections are captured in the planning process and performed in accordance with approved plant procedures. On the basis discussed above, the staff finds that the licensee's ISI program for seals, gaskets and bolted connections provides reasonable assurance that the integrity of the containment pressure boundary will be maintained.

(4) To address the fourth item, the licensee stated, in its May 20 letter, that the only stainless steel expansion bellows associated with the primary containment are on the vent pipes that extend between the drywell and torus. The torus vent pipe expansion bellows are two-ply stainless steel and are designed to be Type B leak rate tested in accordance with 10 CFR Part 50, Appendix J. These expansion bellows are located inside the torus. The cover has cutouts where the leak rate test-taps protrude through. The structural integrity of the bellows was verified by pressurizing them to accident pressure, and venting them from the test-tap on the opposite side. This test method ensured that the two plies were not in contact or restricting the flow. In response to NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," dated March 3, 1992, these bellows were leak rate tested with the test results showing no restriction to

full flow, and leak integrity was verified by measured leakage to be within allowable for all individual bellows.

The licensee also indicated that due to the design of the vent pipe expansion bellows located inside the torus, in order for containment atmosphere to leak through these bellows to the reactor building atmosphere, it would have to pass through both plies of the bellows, which would necessitate a double failure to occur. These bellows are essentially static devices in that they are designed for thermal expansion between the drywell and torus during a design-basis accident, and, therefore, have not experienced the inservice stresses that would propagate transgranular stress corrosion cracking. Based on the design, service conditions and current testing applied to these expansion bellows, additional testing or inspection is not warranted.

On the basis of the above discussion, the item related to NRC Information Notice 92-20 is adequately addressed.

(5) With regard to the item related to the inaccessible areas of the containment liner for which degradations cannot be detected by visual examinations, the licensee, as discussed in its July 28 application, as supplemented, performed an ILRT extension risk assessment considering the potential age related corrosion effects on the containment liner integrity and a series of parametric sensitivity studies. The results of the risk assessment indicated that the ILRT interval extension has a minimal impact on plant risk. From its review of the licensee's submittals, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 and is, therefore, acceptable. The details of the staff's evaluation regarding the risk assessment performed by the licensee is described in Section 3.3 of this safety evaluation.

On the basis of its review of the information provided by the licensee in its TS amendment request and its response to the staff's questions, the staff finds that (1) the structural integrity of the containment vessel is verified through the periodic ISIs conducted as required by Subsection IWE of the ASME Code, Section XI, and (2) the integrity of the penetrations and containment isolation valves are periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J. In addition, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities, if any, in accordance with Article IWE-5000 of the ASME Code, Section XI.

3.3 Risk Assessment

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the July 28, 2003, application. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

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 the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis 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 this basis, industry undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

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 FitzPatrick early in the plant's life required a Type A test frequency of 3 tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to 1 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 pressurized-water reactor and BWR representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 tests in 10 years to 1 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.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking was completed in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per year and increases in large early release frequency (LERF) less than 10^{-7} per year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original frequency of 3 tests in a 10-year interval. 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. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided analyses, as discussed below. The following comparisons of risk from a change in test frequency from 3 tests in 10 years to 1 test in 15 years are considered to be bounding for FitzPatrick comparative frequencies of 1 test in 10 years to 1 test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test interval:

1. Given the change from a 3 in 10-year test frequency to a 1 in 15-year test frequency, the increase in the total integrated plant risk is estimated to be less than 0.01 person-rem per year. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from 3 in 10 years to 1 in 20 years leads to an “imperceptible” increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original 3 in 10 years to 1 in 15 years is estimated to be 2.6×10^{-8} per year based on the internal events PRA. However, there is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in ASME Code, Section XI, Subsections IWE/IWL). Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee’s risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment liner on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 1×10^{-8} per year. The staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of 1.1 percentage points for the cumulative change of going from a test frequency of 3 in 10 years to 1 in 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy, of RG 1.174 and, therefore, is acceptable.

3.4 Summary

On the basis of the technical evaluation of the structural and leak-tightness of the containment and the risk evaluation of the interval extension, the NRC staff finds the proposed one-time extension of the Type A containment ILRT from 10 to no more than 15 years is acceptable. Thus, the proposed change to TS Section 5.5.6 is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (69 FR 44696). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (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.

7.0 REFERENCES:

1. Entergy letter, T. A. Sullivan to NRC, "Proposed License Amendment to Provide a One-Time Integrated Leak Rate test (ILRT) Interval Extension - James A. FitzPatrick Nuclear Power Plant," dated July 28, 2003. (ADAMS Accession No. ML032170128)

2. NRC letter, K. R. Cotton to W. J. Cahill, Jr., Power Authority of the State of New York, "Issuance of Amendment for James A FitzPatrick Nuclear Power Plant," dated October 4, 1996. (ML010960095)

3. Entergy letter, T. A. Sullivan to NRC, "Response to Request for Information Regarding Proposed One Time Deferral of Integrated Containment Leak Rate Testing - James A. FitzPatrick Nuclear Power Plant," dated May 20, 2004. (ML041540407)

4. NRC letter, R. J. Laufer to M. Kansler, Entergy, "James A. FitzPatrick Nuclear Power Plant - Alternative to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) to Use the 1998 Edition of Subsection IWE of the ASME Code, Section XI for Containment Inspections," dated May 1, 2002. (ML020500402)

Principal Contributors: T. Cheng
R. Palla
J. Pulsipher

Date: September 28, 2004