UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

February 19, 2008

MEMORANDUM TO: ACRS Members

FROM: Charles G. Hammer, Senior Staff Engineer **/RA/**
Reactor Safety Branch, ACRS

SUBJECT: TRANSMITTAL OF STATUS REPORT AND PROPOSED SCHEDULE FOR FITZPATRICK LICENSE RENEWAL AT FULL COMMITTEE MEETING ON MARCH 6, 2008

The full Committee will meet on March 6, 2008 and review the FitzPatrick license renewal. To prepare for this meeting, a Status Report and Proposed Schedule are attached.

The final safety evaluation report (SER) for the FitzPatrick license renewal was transmitted to you during the February 7-9, 2008 ACRS meeting. Also, the following material was sent prior to the September 5, 2007 Subcommittee meeting:

1. FitzPatrick License Renewal Application (LRA)
2. Staff's SER with Open Items
3. Staff's onsite Audit Summary Report
4. Region I's LRA Inspection Report Status Report

If you have any questions, please contact either Maitri Banerjee (301) 415-6973 (mxb@nrc.gov) or me at (301) 415-7363 (cgh@nrc.gov).

cc: F. Gillespie
S. Duraiswamy
C. Santos
M. Banerjee
C. Brown

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REVIEW OF LICENSE RENEWAL OF
FITZPATRICK NUCLEAR POWER PLANT
MARCH 6, 2008
ROCKVILLE, MARYLAND

- STATUS REPORT -

PURPOSE

The purpose of this session is to review the License Renewal Application (LRA) for James A. FitzPatrick Nuclear Power Plant (JAFNPP) and the associated final Safety Evaluation Report (SER) dated February 2008. The full Committee will hear presentations by and hold discussions with representatives of the staff and Entergy Nuclear Operations, Inc. The License Renewal Subcommittee previously met to discuss this application and the draft SER with open items on September 5, 2007.

BACKGROUND

JAFNPP is located approximately seven miles northeast of Oswego, New York. The NRC issued the JAFNPP construction permit on May 20, 1970, and the operating license on October 17, 1974. JAFNPP is of a Mark 1, GE 4, boiling water reactor design. GE supplied the nuclear steam supply system and Stone and Webster originally designed and constructed the balance of the plant. The JAFNPP licensed power output is 2536 megawatts thermal with a gross electrical output of approximately 881 megawatts electric. The current facility operating license for JAFNPP expires at midnight October 17, 2014.

DISCUSSION

By letter dated July 31, 2006, Entergy Nuclear Operations, Inc. (ENO or the applicant) submitted the LRA (Reference 1) in accordance with Title 10, Part 54, of the *Code of Federal Regulations*, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." ENO requests renewal of the JAFNPP operating license for an additional 20 years. The NRC staff has prepared a Safety Evaluation Report (Reference 2) to summarize the results of its safety review of the LRA for compliance with Title 10, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," of the *Code of Federal Regulations* (10 CFR Part 54) and the guidance in the NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), dated September 2005.

The applicant stated that it had not identified any Technical Specification (TS) changes necessary to support issuance of the renewed operating license.

The staff used the following Interim Staff Guidance (ISG) in the FitzPatrick LRA review: Nickel-alloy components in the reactor coolant pressure boundary **(LR-ISG-19B)** and Corrosion of drywell shell in Mark I containments **(LR-ISG-2006-01)**.

The SER presents the status of the staff's review of the FitzPatrick LRA. The SER discusses two open items which were open at the time of the September 5, 2007 Subcommittee meeting. As discussed below, the staff has subsequently closed these OIs. The SER contains: no confirmatory items which must be resolved before the staff can make a final determination on the LRA, three proposed license conditions, and 25 commitments.

SUMMARY OF OPEN ITEMS

As a result of its review of the LRA, including additional information submitted through June 20, 2007, the staff had earlier identified the following two open items (OIs). An item was considered open if, in the staff's judgement it did not meet all applicable regulatory requirements at the time of the issuance of the SER with Open Items. The staff assigned a unique identifying number to each OI. As discussed below, all OIs have been closed by the staff.

OI 4.2.1-1: (SER Section 4.2.1 - Reactor Vessel Neutron Fluence)

The staff reviewed GE-NE-B1100732-01 report on analysis of the 120^0 capsule removed at 13.4 effective full power years (EFPYs) of operation submitted by the applicant to determine if calculation of fluence values were in accordance with the guidance of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The staff determined that the GE-NE-B1100732-01 report did not conform to RG 1.190. The applicant had stated at the time that it would submit a new fluence calculation to the staff for review when complete. This item was identified as OI 4.2.1-1 in the staff SER With Open Items issued on July 31, 2007.

The applicant, in its November 5, 2007, response to OI 4.2.1-1, submitted a summary of a new analysis for its determination of maximum pressure vessel neutron fluence. The staff reviewed the applicant's new calculated values and found the new calculation acceptable as it adhered with the guidance of RG 1.190. The staff had determined that the original values that were submitted in LRA remain bounded by the new calculated values; therefore, OI 4.2.1-1 is closed. The staff's evaluation of this item is detailed in SER Section 4.0.

As noted in the staff's SER with Open Items issued July 31, 2007, for OI 4.2.1-1, the applicant's reactor pressure vessel neutron fluence evaluation for the period of extended operation impacted the staff's review of other items in LRA Section 4.2.2.

As stated in the staff's SER with Open Items issued July 31, 2007, the staff had found information submitted for the following TLAA sections of the LRA acceptable pending resolution of OI 4.2.1-1. Now that OI 4.2.1-1 is resolved and closed, the staff finds the following sub-OIs (sOIs) closed:

> **sOI 4.2.2-1:** (SER Section 4.2.2 - Pressure-Temperature Limits)
>
> The staff's review of P-T limits was based on the applicant's original fluence values in LRA Section 4.2.1. These original values had been previously found acceptable as documented in the previously issued SER with Open Items dated July 31, 2007. The staff finds that these original values remain bounded by the new calculated values submitted by the applicant in a letter dated November 5, 2007. Therefore, sOI 4.2.2-1 is now closed.
>
> The staff determined that the TLAA for P-T limits is acceptable because: (1) the projected 54 EFPY fluence and ART values are in fact less than the 200 ^{0}F suggested in RG 1.99, Section 3, and (2) changes to the P-T limit curves will be implemented by the license amendment process (i.e., through revisions of the plant TS) in accordance with 10 CFR 50.60 and 10 CFR Part 50, Appendix G.
>
> **sOI 4.2.3-1:** (SER Section 4.2.3 - Charpy Upper-Shelf Energy)

The staff determined that the applicant correctly used RG 1.99, Revision 2, Position 1 to calculate the predicted percentage decrease in upper-shelf energy (USE) conservatively for the period of extended operation. The staff also independently calculated: (1) the end of life (EOL) USE values for the beltline plate materials at 54 EFPY and (2) the equivalent margin analysis (EMA) of the percent drop in USE for the beltline weld materials through 54 EFPY. Verifying the drop in USE values from neutron irradiation using the RG 1.99, Revision 2 methodology, the staff finds that all the beltline materials meet 10 CFR Part 50, Appendix G, EOL USE or EMA requirements and SRP-LR Section 4.2.3.1.1.2 criteria for USE/EMA TLAAs in accordance with 10 CFR 54.21(c)(1)(ii). SER Table 4.2.3-1 summarizes the results of both the applicant's and the staff's independent USE/EMA calculations for the limiting plate and weld materials.

The staff's review was based on the fluence values provided by the applicant in LRA Section 4.2.1. The staff finds that the values that were submitted by the applicant's LRA submitted on August 1, 2006, remains bounded by the new calculated values submitted by the applicant in a letter dated November 5, 2007. Therefore, sOI 4.2.3-1 is now closed.

sOI 4.2.4-1: (SER Section 4.2.4 - Adjusted Reference Temperature)

The staff confirmed that lower shell axial welds 2-233 A, B, and C fabricated from Heat No. 27204/12008 were the limiting 1/4T reference temperature (nil-ductility transition) RT_{NDT} reactor vessel components. The staff calculated a limiting 1/4T RT value of 132.1 ^{0}F for this plate material based on the chemistry factor (CF) table for plate/forging materials in RG 1.99, Revision 2 and a 1/4T fluence of 0.174 x 10^{19} n/cm (E > 1.0 MeV) at 54 EFPY. The 1/4T RT_{NDT} value calculated by the staff at 54 EFPY is within 3.2 ^{0}F of that calculated (i.e., 135.3 ^{0}F) by the applicant for this material. As the staff's independent 1/4T RT_{NDT} value agreed with that calculated by the applicant, the staff found the applicant's calculated and projected limiting 1/4T RT_{NDT} value for the reactor vessel at 54 EFPY valid and found the TLAA on 1/4T RT_{NDT} values for the reactor vessel through 54 EFPY acceptable in accordance with 10 CFR 54.21(c)(1)(ii).

The staff's review was based on the applicant's fluence values in LRA Section 4.2.1. The staff finds that the values that were submitted by the applicant's LRA on August 1, 2006, remain bounded by the new calculated values submitted by the applicant in a letter dated November 5, 2007. Therefore, sOI 4.2.4-1 is now closed.

sOI 4.2.5-1: (SER Section 4.2.5 - Reactor Vessel Circumferential Weld Inspection Relief)

The staff finds the applicant's evaluation for this TLAA acceptable because the 54 EFPY conditional failure probability for the reactor vessel circumferential welds is bounded by the analysis in the staff SER dated July 28, 1998, and the applicant will use procedures and training to limit cold over-pressure events during the period of extended operation. This analysis satisfies the evaluation requirements of the staff SER dated July 28, 1998; however, the applicant still must request relief from the circumferential weld examination for the period of extended operation in accordance with 10 CFR 50.55a.

The staff's review was based on the applicant's fluence values in LRA Section 4.2.1. The staff finds that the values that were submitted by the applicant's LRA submitted on August 1, 2006, remains bounded by the new calculated values submitted by the applicant in a letter dated November 5, 2007. Therefore, sOI 4.2.5-1 is now closed.

sOI 4.2.6-1: (See SER Section 4.2.6 - Reactor Vessel Axial Weld Failure Probability)

The staff reviewed LRA Section 4.2.6, to verify pursuant to 10 CFR 54.21(c)(1)(ii) that the analyses have been projected to the end of the period of extended operation. The staff reviewed the applicant's TLAA of the reactor vessel axial weld failure probability, as summarized in LRA Section 4.2.6, and its response to RAI 4.2.6.1 dated February 12, 2007, supplemented by letter dated June 20, 2007, and determines that the applicant appropriately described how the conditional failure probability for the reactor vessel axial welds is bounded by the analysis in the staff supplemental SER dated March 7, 2000, on the BWRVIP-05 Report for the period of extended operation. The staff therefore finds the applicant's TLAA Section 4.2.6 and UFSAR supplement summary description A.2.2.1.6 acceptable. The staff concludes that the applicant's TLAA for the reactor vessel axial weld failure probability comply with the 10 CFR 54.21(c)(1)(ii) TLAA acceptance criterion.

The staff's review was based on the applicant's fluence values in LRA Section 4.2.1. The staff finds that the new calculated value submitted by the applicant in a letter dated November 5, 2007 remains bounded by the value that was submitted by the applicant's LRA submitted on August 1, 2006. Therefore, sOI 4.2.6-1 is now closed.

sOI B.1.24-3: (See SER Section 3.0.3.2.16 - Reactor Vessel Surveillance Program)

On the basis of the staff's review for LRA item B.1.24 discussed in SER Section 3.0 and the new calculated fluence value submitted by the applicant in a letter dated November 5, 2007, the staff finds that the values that were submitted by the applicant's LRA submitted on August 1, 2006, remains bounded by the new calculated values submitted by the applicant in a letter dated November 5, 2007. Therefore, the applicant's implementation of the Integrated Surveillance Program (ISP), as specified in the BWRVIP-116 Report, remains valid and as such, the various attributes in the ISP are not affected by the new methodology of calculating the neutron fluence values. The staff finds that the applicant has demonstrated that the effects of aging due to loss of fracture toughness of the reactor pressure vessel beltline region will be adequately managed, so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). Therefore, sOI B.1.24-3 is now closed.

OI 4.3.3-1: (SER Section 4.3.3 - Effects of Reactor Water Environment on Fatigue Life)

By letter dated August 14, 2007, as supplemented by letter dated November 5, 2007, the applicant responded to RAI 4.3.3-1 and resolved the staff's issue identified in OI 4.3.3-1.

In these letters, the applicant amended the LRA and supplemented Commitment No. 20 to justify its environmentally-assisted fatigue analysis. The November 5, 2007, letter clarified that Option 1 of Commitment No. 20 for refined CUF calculations is consistent with NRC recommendations for periodic CUF updates in "monitoring and trending" (i.e., program element 4) of GALL AMP X.M1, "Metal Fatigue of the Reactor Coolant Pressure Boundary," and for "corrective actions" in GALL AMP X.M1. The applicant also clarified that Options 2 and 3 of Commitment No. 20 are corrective actions consistent with those recommended in "corrective action" (i.e., program element 7) of the same GALL AMP. With these clarifications, the applicant amended the LRA to bring Commitment No. 20 within the scope of the Fatigue Monitoring Program and to credit

this AMP as the basis for acceptance of this TLAA as in accordance with 10 CFR 54.21(c)(1)(iii).

After reviewing the letter dated August 14, 2007, as supplemented by the letter dated November 5, 2007, the staff finds the applicant's clarification of these changes consistent with NRC recommendations in GALL AMP X.M1 and therefore are acceptable. The staff concludes that the applicant's response is acceptable and, therefore, OI 4.3.3.1 is closed. The staff's evaluation of the applicant's response is detailed in SER Section 4.3.3.

SUMMARY OF CONFIRMATORY ITEMS

There are no confirmatory items in the SER.

SUMMARY OF PROPOSED LICENSE CONDITIONS

Following the staff's review of the LRA, including subsequent information and clarifications from the applicant, the staff identified three proposed license conditions.

The first license condition requires the applicant to include the UFSAR supplement required by 10 CFR 54.21(d) in the next UFSAR update required by 10 CFR 50.71(e) following the issuance of the renewed license.

The second license condition requires future activities described in the UFSAR supplement to be completed prior to the period of extended operation.

The third license condition requires that all capsules in the reactor vessel that are removed and tested meet the requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the staff prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the staff, as required by 10 CFR Part 50, Appendix H.

SUMMARY OF COMMITMENTS

During the review by the NRC staff, the applicant made commitments which are listed in detail in Appendix A to the SER. The applicant made 25 commitments related to the AMPs to manage aging effects of structures and components. The following is a summary list of these commitments:

1. Implement the Buried Piping and Tanks Inspection Program as described in LRA Section B.1.1

2. Enhance the BWR CRD Return Line Nozzle Program to examine the CRDRL nozzle-to-vessel weld and the CRDRL nozzle inside radius section per Section XI Table IWB-2500-1 Category B-D Items B3.90 and B3.100.

3. Enhance the Diesel Fuel Monitoring Program to include periodic draining, cleaning, visual inspections, and ultrasonic measurement of the bottom surfaces of the fire pump diesel fuel oil tanks, EDG day tanks, and EDG fuel oil storage tanks to ensure that significant degradation is not occurring. Enhance the Diesel Fuel Monitoring Program to

specify acceptance criteria for UT measurements of diesel generator fuel storage tanks within the scope of this program.

4. Enhance the External Surfaces Monitoring Program guidance documents to include periodic inspections of systems in scope and subject to aging management review (AMR) for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).

5. Enhance the Fire Protection Program to inspect fire barrier walls, ceilings, and floors at least once every refueling outage. Inspection results will be acceptable if there are no visual indications of degradation such as cracks, holes, spalling, or gouges. Enhance the Fire Protection Program to inspect at least one seal of each type every 24 months.

6. Enhance the Fire Water Program to include inspection of hose reels for corrosion. Acceptance criteria will be enhanced to verify no significant corrosion. Enhance Fire Water Program to include visual inspection of spray and sprinkler system internals for evidence of corrosion. Acceptance criteria will be enhanced to verify no significant corrosion. Enhance Fire Water Program to include that a sample of sprinkler heads will be inspected using guidance of NFPA 25 (2002 Edition) Section 5.3.1.1.1. NFPA 25 also contains guidance to repeat sampling every 10 years after initial field service testing. Enhance Fire Water Program to include that wall thickness evaluations of fire water piping will be performed on system components using non-intrusive techniques to identify evidence of loss of material due to corrosion. These inspections will be performed before the end of the current operating term and at intervals thereafter during the period of extended operation. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function.

7. Implement the Heat Exchanger Monitoring Program as described in LRA Section B.1.15.

8. Implement the Metal-Enclosed Bus Inspection Program as described in LRA Section B.1.17.

9. Implement the Non-EQ Instrumentation Circuits Review Program as described in LRA Section B.1.18.

10. Implement the Non-EQ Insulated Cables and Connections Program as described in LRA Section B.1.19.

11. Enhance the Oil Analysis Program to periodically sample lubricating oil in the security generator, and the fire pump diesel. Enhance the Oil Analysis Program to include viscosity and neutralization number determination of oil samples from components that do not have regular oil changes. Enhance the Oil Analysis Program to include particulate and water content for oil replaced periodically.

12. Implement the One-Time Inspection Program as described in LRA Section B.1.21.

13. Enhance the Periodic Surveillance and Preventive Maintenance Program as necessary to assure that the effects of aging will be managed in accordance with JAF-RPT-05-LRD02.

14. Enhance the Reactor Vessel Surveillance Program to include the data analysis, acceptance criteria, and corrective actions described in LRA Section B.1.24.

15. Implement the Selective Leaching Program in accordance with the program as described in LRA Section B.1.25.

16. Enhance the Structures Monitoring Program procedure to:

    - specify that manholes, duct banks, underground fuel oil tank foundations, manway seals and gaskets, hatch seals and gaskets, underwater concrete in the intake structure, and crane rails and girders are included in the program.

    - include guidance for performing structural examinations of elastomers and rubber components to identify cracking and change in material properties.

    - include guidance for performing periodic inspections to confirm the absence of aging effects for lubrite surfaces in the torus radial beam seats and for lubrite surfaces in the torus support saddles.

    - perform an engineering evaluation on a periodic basis (at least once every five years) of groundwater samples to assess aggressiveness (pH < 5.5, chloride > 500 ppm and sulfate > 1500) of groundwater to concrete.

    - inspect any inaccessible concrete areas that may be exposed by excavation for any reason, or any inaccessible area where observed conditions in accessible areas, which are exposed to the same environment, show that significant concrete degradation is occurring.

17. Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Section B.1.28.

18. Enhance the Water Chemistry Control - Auxiliary Systems Program to include guidance for sampling the control room and relay room chilled water, decay heat removal cooling water, and the security generator jacket cooling water.

19. Enhance the Bolting Integrity Program to include guidance from EPRI NP-5769 and EPRI TR-104213. Enhance the Bolting Integrity Program to clarify that actual yield strength is used in selecting materials for low susceptibility to SCC and to clarify the prohibition on use of lubricants containing MoS2 for bolting.

20. At least 2 years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for BWRs of the JAFNPP vintage, JAFNPP will implement one or more of the following:

    (1) Refine the fatigue analyses or develop new analyses (Class 1 RHR piping and Class 1 feedwater piping locations), if necessary, to determine valid CUFs less than 1 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following options.

- For locations, including NUREG/CR-6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF.

- More limiting JAFNPP-specific locations with a valid CUF may be added in addition to the NUREG/CR-6260 locations.

- Representative CUF values from other plants, adjusted to or enveloping the JAFNPP plant specific external loads may be used if demonstrated applicable to JAFNPP.

- For locations, including NUREG/CR-6260 locations, an analysis using the NRC-approved ASME code 2001 edition up to and including 2003 addendum, may be performed to determine a valid CUF.

The determination of F_{en} will account for operating time with normal water chemistry and operating time with hydrogen water chemistry.

(2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).

(3) Repair or replace the affected locations before exceeding a CUF of 1.0.

Should JAFNPP select Option 2 to manage environmentally assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.

21. Enhance the BWR Vessel Internals Program to inspect fifteen (15) percent of the top guide locations using enhanced visual inspection techniques. EVT-1, within the first 18 years of the period of extended operation, with at least one-third of the inspections to be completed within the first six (6) years and at least two-thirds within the first 12 years of the period of extended operations. Locations selected for examination will be areas that have exceeded the neutron fluence threshold.

22. Enhance the BWR Vessel Internals Program to ensure the effects of aging on the steam dryer are managed in accordance with the guidelines of BWRVIP-139 as approved by the NRC and accepted by the BWRVIP Executive Committee.

23. Enhance the BWR Vessel Internals Program to perform inspections of the core plate rim hold down bolts.

Appendix A.2.2.7 Core Plate is revised to add that JAFNPP will perform one of the following:

(1) Install core plate wedges prior to the period of extended operation; or,

(2) Complete a plant-specific analysis to determine acceptance criteria for continued inspection of core plate rim hold down bolting in accordance with BWRVIP-25 and submit the inspection plan, along with the acceptance criteria and justification for the

inspection plan, to the NRC two years prior to the period of extended operation for NRC review and approval.

If Option 2 is selected, the analysis to determine acceptance criteria will address the information requested in RAIs 3.1.2-2A and 4.7.3.2-1.

24. Implement the Bolted Connections Program as described in LRA Section B.1.31.

25. Implement the Oil-Filled Cable System aging management that will be controlled by the following programs:

    - External Surfaces Monitoring Program
    - Oil Analysis Program
    - Periodic Surveillance and Preventive Maintenance Program

ONSITE AUDIT AND REGIONAL INSPECTION ACTIVITIES

In support of the staff's review of the LRA for JAFNPP, an NRC project team conducted three onsite audits to review the AMPs, AMRs, and TLAAs and issued a report dated June 19, 2007 (Reference 3). Also, an inspection was performed by Region I which reviewed the screening and scoping of non-safety related systems, structures, and components in AMPs. The Region I inspection report is dated August 2, 2007 (Reference 4).

EXPECTED COMMITTEE ACTION

The Committee will review this matter and may provide a report during the March 6-8, 2008 ACRS meeting.

References

1. ENO License Renewal Application for FitzPatrick, dated July 31, 2006, ML062160491
2. NRC Safety Evaluation Report, dated February 2008, ML080250372
3. NRC Staff onsite Audit Report, dated June 19, 2007, ML071580049
4. NRC Inspection Report 05000333/2007007, dated August 2, 2007, ML072140637

Advisory Committee on Reactor Safeguards
Review of License Renewal of
FitzPatrick Nuclear Power Plant
March 6, 2007
Rockville, MD

-PROPOSED SCHEDULE-

Cognizant Staff Engineer: Charles G. Hammer cgh@nrc.gov (301) 415-7363

Topics	Presenters	Time
Opening Remarks	M. Bonaca, ACRS	8:35 am - 8:40 am
Staff Introduction	P.T. Kuo, NRR	8:40 am - 8:45 am
Entergy Introduction	Entergy	8:45 am - 8:50 am
FitzPatrick License Renewal Application Presentation	Entergy	8:50 am - 9:50 am
NRC Staff Review Summary NRC Onsite Inspection Results NRC Audit	T. Le, NRR G. Myer, Region I K. Chang, NRR	9:50 am - 10:20 am
Committee Discussion	M. Bonaca, ACRS	10:20 am - 10:30 am

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.

- 50 copies of the presentation materials to be provided.