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

August 17, 2007

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 LICENSEE RENEWAL SUBCOMMITTEE MEETING

The Licensee Renewal Subcommittee will meet on September 5, 2007 to review the FitzPatrick license renewal.

To prepare for this meeting, a Status Report and Proposed Schedule are attached.

The following review materials were transmitted to you on August 6, 2007:

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 me at (301) 415-7363 or cgh@nrc.gov.

cc: F. Gillespie
S. Duraiswamy
C. Santos

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE ON PLANT LICENSE RENEWAL
FITZPATRICK NUCLEAR POWER PLANT
SEPTEMBER 5, 2007
ROCKVILLE, MARYLAND

- STATUS REPORT -

PURPOSE

The purpose of this meeting is to review the License Renewal Application (LRA) for James A. FitzPatrick Nuclear Power Plant (JAFNPP), and the associated Draft Safety Evaluation Report (SER) with Confirmatory Items dated July 2007. The Subcommittee will hear presentations by and hold discussions with representatives of the staff and Entergy Nuclear Operations, Inc.

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, 2007, 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 draft Safety Evaluation Report with open items (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 licensee 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 with open items presents the status of the staff’s review of information submitted through June 20, 2007, the cutoff date for consideration in the SER. The SER contains two open items which the staff determined do not meet all applicable regulatory requirements at the time of issuance of the SER, 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. The staff will present its final conclusion on the LRA review in an update to this SER.

SUMMARY OF OPEN ITEMS

As a result of its review of the LRA, including additional information submitted through June 20, 2007, the staff identified the following two open items (OIs). An item is considered open if, in the staff's judgement it does not meet all applicable regulatory requirements at the time of the issuance of the SER with open items. The staff has assigned a unique identifying number to each OI.

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

The staff reviewed GE-NE-B1100732-01 report on analysis of the 120° capsule removed at 13.4 effective full power years (EFPYs) of operation submitted by the applicant to confirm if calculation of fluence values are in accordance with the guidance of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

The staff's review of this report found several discrepancies with the RG 1.190 guidance. For determining pressure vessel neutron fluence, the staff finds the projected fluence values unacceptable. The applicant has stated that it will submit a new fluence calculation by a contractor to the staff for review when complete. This item has been identified as **OI 4.2.1-1**.

As noted in OI 4.2.1-1, the applicant's reactor pressure vessel neutron fluence evaluation for the period of extended operation remains an issue to be resolved. This fluence evaluation impacts the staff review of LRA Section 4.2.2 to verify accordance with 10 CFR 54.21(c)(1)(ii) and projection of the analyses to the end of the period of extended operation.

The staff has determined that this OI affects the acceptability of several Time-Limited Aging Analysis (TLAA) sections and one AMP section submitted in the LRA. Thus these items are identified as sub-OIs (**sOIs**) of OI 4.2.1-1, and are listed below:

> **sOI 4.2.2-1:** (SER Section 4.2.2 - Pressure-Temperature Limits)
>
> The staff finds the applicant's plan to manage the pressure-temperature (P-T) limits acceptable because it will implement changes to the P-T limit curves by the license amendment process (*i.e.*, through revisions of the plant technical specification (TS)) and will meet the requirements of 10 CFR 50.60, 10 CFR Part 50, Appendix G, and the TLAA acceptance guidance of SRP-LR Section 4.2.2.1.3.3.
>
> The staff's review of P-T limits was based on the applicant's fluence values in LRA Section 4.2.1. Until OI 4.2.1-1 is resolved, the staff cannot close its review of this TLAA. This item is **sOI 4.2.2-1.**
>
> **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. Until OI 4.2.1-1 is resolved, the staff cannot close its review of this TLAA. This item is **sOI 4.2.3-1.**

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_{NDT} value of 132.1 °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^2 (E > 1.0 MeV) at 54 EFPY. The 1/4T RT_{NDT} value calculated by the staff at 54 EFPY is within 3.2 °F of that calculated (i.e., 135.3 °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. Until OI 4.2.1-1 is resolved, the staff cannot close its review of this TLAA. This is **sOI 4.2.4-1.**

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. Until OI 4.2.1-1 is resolved, the staff cannot close its review of this TLAA. This item is **sOI 4.2.5-1**.

(Note that SER Section 4.2.5 discusses a "CI 4.2.5-1" in error. There are no confirmatory items (CIs) in the SER. The staff has verbally verified that the correct reference should be to OI 4.2.1-1.)

<u>sOI 4.2.6-1:</u> (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 pending resolution of OI 4.2.1-1. The staff concludes that the applicant's TLAA Section 4.2.6 and UFSAR supplement A.2.2.1.6 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. Until OI 4.2.1-1 is resolved, the staff cannot close its review of this TLAA. This item is **sOI 4.2.6-1**.

(Note that SER Section 4.2.6 discusses a "CI 4.2.6-1" in error. There are no confirmatory items (CIs) in the SER. The staff has verbally verified that the correct reference should be to OI 4.2.1-1.)

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

On the basis of the staff review for LRA item B.1.24 discussed in SER Section 3.0, the staff finds, pending the resolution of OI 4.2.1-1, 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). The staff has identified this as **sOI B.1.24-3** for OI 4.2.1-1.

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

The staff noted that under Commitment No. 20 the applicant must either (1) redo the 60-year environmentally-adjusted cumulative usage factor (CUF) calculations for the Class 1 locations in LRA Table 4.3-3, including new environmental CUF calculations for the Class 1 portions of the residual heat removal and feedwater piping, (2) manage the aging effects due to fatigue by using an staff-approved Aging Management Program (AMP) to inspect these component locations prior to the period of extended operation, or (3) repair or replace the affected components before exceeding a environmentally-adjusted CUF value of 1.0. If using the first option as the basis for acceptance of the TLAA on environmental fatigue, the applicant will submit the results of the environmentally-adjusted CUF calculations for review and approval at least two years prior to the period of extended operation.

The staff concludes that the Fatigue Monitoring Program, when supplemented by Commitment No. 20, provides an acceptable basis for managing the impact of the reactor coolant system environment on the Class 1 fatigue calculations, as evaluated in accordance with 10 CFR 54.21(c)(1)(iii), because the applicant either (1) will recalculate new 60-year, environmentally-adjusted CUF values to confirm whether the revised environmentally-adjusted CUFs are less than 1.0 or (2) will manage the effects of fatigue on the components by inspecting for fatigue-induced cracking and with a staff-approved AMP, or (3) will repair or replace the components prior to the period of extended operation.

In LRA Amendment No. 12, June 20, 2007 (ML071770168), the applicant amended the LRA to change the basis for accepting this TLAA in accordance with the criteria of 10 CFR 54.21(c)(1)(iii).

In RAI 4.3.3-1 dated July 25, 2007 (ML072010267), the staff sought further clarifications on the options that could be used under LRA Commitment No. 20 and asked the applicant to identify which option or options under LRA Commitment No. 20 would be used to satisfy the commitment when implemented and, for each option selected to meet the commitment, to provide a sufficient detailed description of the methodology that would be used to satisfy the option. The staff informed the applicant that the information requested in the RAI was necessary in order for the staff to make a determination on the acceptability of the applicant's TLAA on environmentally-assisted fatigue. The staff therefore requested that the information in the response to RAI 4.3.3-1 be submitted as an amendment to the LRA. The specific details of RAI 4.3.3-1 are provided in ADAMS Accession No. ML072010267.

The staff's determination on the acceptability of the TLAA on environmentally-assisted fatigue is pending submittal of the applicant's response to RAI 4.3.3-1 and the staff's review of the response to this RAI. **This is OI 4.3.3-1.**

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 licensee made commitments which are listed in detail in Appendix A to the SER. The licensee 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 SUBCOMMITTEE ACTION

The Subcommittee Chairman will provide a report to the full Committee during the September 2007 ACRS meeting.

References

1. ENO License Renewal Application for FitzPatrick, dated July 31, 2006.
2. NRC Safety Evaluation Report with Open Items, dated July 2007.
3. NRC Staff onsite Audit Report, dated June 19, 2007.
4. NRC Inspection Report 05000333/2007007, dated August 2, 2007.

Advisory Committee on Reactor Safeguards
Plant License Renewal Subcommittee Meeting
FitzPatrick Nuclear Power Plant
September 5, 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	10:30 pm - 10:35 pm
Staff Introduction	P.T. Kuo, NRR	10:35 pm - 10:40 pm
FitzPatrick License Renewal Application A. Application Background B. Description of FitzPatrick C. Operating History D. Scoping Discussion E. Application of GALL F. Commitment Process	Entergy Nuclear FitzPatrick	10:40 pm - 12:00 pm
Lunch Break		12:00 pm - 1:00 pm
FitzPatrick License Renewal Application (Con't)	Entergy Nuclear FitzPatrick	1:00 pm - 2:00 pm
SER Overview A. Scoping and Screening Results B. Onsite Inspection Results	NRR - T. Le Region I - R. Conte G. Myer	2:00 pm - 2:30 pm
Aging Management Program Review and Audits	NRR - T. Le J. Medoff	2:30 pm - 3:00 pm
Break		3:00 pm - 3:15 pm
Time-Limited Aging Analyses	NRR - T. Le J. Medoff	3:15 pm - 4:15 pm
Subcommittee Discussion	M. Bonaca, ACRS	4:15 pm - 5:00 pm

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.