The FEIS describes the proposed action and alternatives to the proposed action, including the no-action alternative. The FEIS assesses the impacts of the proposed action and its alternatives on human health, air quality, water resources, waste management, geology, noise, ecology, land use, cultural resources, socioeconomics, accident impacts, and environmental justice. Additionally, the FEIS analyzes and compares the costs and benefits of the proposed action.

After weighing the impacts, costs, and benefits of the proposed action and comparing alternatives (*see* Sections 2.6, 4.15, and 7 of the FEIS), the NRC staff, in accordance with 10 CFR 51.91 (d), sets forth its final NEPA recommendation regarding the proposed action. The NRC staff recommend that, unless safety issues mandate otherwise, the action called for is the issuance of the proposed license to FWENC. In this regard, the NRC staff concludes (i) the applicable environmental monitoring program described in Section 6 of the FEIS, and (ii) the proposed mitigation measures discussed in Section 5 of the FEIS would eliminate or substantially lessen any potential adverse environmental impacts associated with the proposed action.

The NRC staff has concluded that the overall benefits of the proposed Idaho Spent Fuel Facility outweigh the disadvantages and costs, based on consideration of the following:

—The proposed Idaho Spent Fuel Facility will have small impacts on the physical environment and human communities in the vicinity. Long-term impacts of the no-action alternative are likely to be similar to the impacts of the proposed action.

—The proposed action is designed to support the INEEL mission and comply with agreements and commitments negotiated by DOE, including the 1995 Settlement Agreement among DOE, the State of Idaho, and the U.S. Navy to remove SNF from Idaho by 2035.

—Currently, most of the SNF to be received by the proposed Idaho Spent Fuel Facility is stored at the Idaho Nuclear Technology Center. Transfer distances from current storage locations to the proposed facility are relatively short.

—The current storage configuration does not prepare the SNF for shipment from INEEL to a national HLW repository.

NRC staff in the Spent Fuel Project Office are currently completing the licensing and safety review of FWENC's proposed action. The final licensing decision is currently scheduled for the Spring of 2004.

Dated at Rockville, Maryland, this 3rd day of February 2004.

For the Nuclear Regulatory Commission.

Lawrence E. Kokajko,

Chief, Environmental and Performance Assessment Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E4–413 Filed 2–26–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment; Notice of Meeting

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment will hold a meeting on March 25, 2004, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, March 25, 2004—1 p.m. Until the Conclusion of Business

The purpose of this meeting is to discuss the NRC staff's draft action plan for the implementation of the phased approach to PRA Quality. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Michael R. Snodderly (telephone: 301–415–6927) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted during the meeting.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: February 23, 2004.

Sam Duraiswamy,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. E4–414 Filed 2–26–04; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[Docket Nos. (as shown in the Attachment), License Nos. (as shown in the Attachment), EA–03–009]

In the Matter of All Pressurized Water Reactor Licensees; First Revised Order Modifying Licenses

I

The Licensees identified in the Attachment to this Order hold licenses issued by the Nuclear Regulatory Commission (NRC or Commission) authorizing operation of pressurized water reactor (PWR) nuclear power plants in accordance with the Atomic Energy Act of 1954 and title 10 of the Code of Federal Regulations (10 CFR) part 50.

II

The reactor pressure vessel (RPV) heads of PWRs have penetrations for control rod drive mechanisms and instrumentation systems. Nickel-based alloys (*e.g.*, Alloy 600) are used in the penetration nozzles and related welds. Primary coolant water and the operating conditions of PWR plants can cause cracking of these nickel-based alloys through a process called primary water stress corrosion cracking (PWSCC). The susceptibility of RPV head penetrations to PWSCC appears to be strongly linked to the operating time and temperature of the RPV head. Problems related to PWSCC have, therefore, increased as plants have operated for longer periods of time. Inspections of the RPV head nozzles at the Oconee Nuclear Station, Units 2 and 3 (Oconee), in early 2001 identified circumferential cracking of the nozzles above the J-groove weld, which joins the nozzle to the RPV head. Circumferential cracking above the J-groove weld is a safety concern because of the possibility of a nozzle ejection if the circumferential cracking is not detected and repaired.

Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), which is incorporated into NRC regulations by 10 CFR 50.55a, "Codes and standards,'' currently specifies that inspections of the RPV head need only include a visual check for leakage on the insulated surface or surrounding area. These inspections may not detect small