UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 29, 2005

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 523rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, June 1-3, 2005 AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 523rd meeting, June 1-3, 2005, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report, letters, and memoranda:

REPORT:

Report to Nils J. Diaz, Chairman, NRC, from Graham B. Wallis, Chairman, ACRS:

- Draft Commission Paper on "Risk-Informed Alternatives to the Single Failure Criterion," dated June 10, 2005

LETTERS:

Letters to Luis A. Reyes, Executive Director for Operations, NRC, from Graham B. Wallis, Chairman, ACRS:

- Interim Report on the Safety Aspects of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2, dated June 9, 2005

- Draft Final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," dated June 10, 2005

- Interim Letter: Draft Safety Evaluation Report on Grand Gulf Early Site Permit Application, dated June 14, 2005

- Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," dated June 10, 2005

MEMORANDA:

Memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Proposed Bulletin Entitled, "Emergency Preparedness and Response Actions for Security-Based Events," dated June 3, 2005
- Draft Final Regulatory Guides on ASME Code Cases, dated June 7, 2005

HIGHLIGHTS OF KEY ISSUES

1. Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2

The Committee met with NRC staff and representatives of the Nuclear Management Company, LLC to perform an interim review of the license renewal application of the Point Beach Nuclear Plant, Units 1 and 2 and the related Safety Evaluation Report (SER) with open items. The operating licenses for Units 1 and 2 expire on October 5, 2010, and March 8, 2013, respectively. The applicant has requested approval for continued operation of each unit for a period of 20 years beyond the current license expiration dates. The applicant presented its plan for sustaining improvement at the plant. The staff described the "Red" findings and the Confirmatory Action Letter (CAL) issued to the Point Beach Nuclear Power Plant. The five areas of concern outlined in the CAL are human performance, engineering design control, engineering/operations interface, emergency preparedness, and the Corrective Action Program. The staff stated that the applicant has made progress in all five areas, but the CAL remains open.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations, dated June 9, 2005, describing its concerns with the applicant's corrective action program and human performance that may affect the development and effective implementation of license renewal programs. The Committee recognizes that current operating performance is not a consideration in the license renewal rule, but is concerned that aspects of current performance may affect the development and effective implementation of license renewal programs. During its review of the final SER, the Committee would like to hear what additional actions the staff plans to take to increase confidence that the requirements of the license renewal rule have been met and there is reasonable assurance that aging degradation can be adequately managed.

2. Draft Commission Paper on Policy Issues Related to New Plant Licensing

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the draft Commission paper on policy issues related to new plant licensing. The Committee was briefed on two policy issues: the minimum level of safety that new plants need

to meet to achieve enhanced safety; and how to account for the risk from multiple reactors at a single site.

The staff has defined four options for the level of safety issue. The staff recommends the option that would establish the Quantitative Health Objectives (QHOs) as the minimum level of safety and the regulatory framework will be designed to meet this level of safety. Such an approach goes beyond current practice in that compliance with the QHOs would be mandatory.

For the integrated risk issue, the staff has defined three options. The staff recommends the option that the integrated risk from all new units at the site satisfy the Commission's QHOs.

Committee Action

The Committee plans to continue its debate on these two policy issues during its July 6-8, 2005 meeting.

3. Fire Risk Requantification and Probabilistic Risk Assessment (PRA) Methodology for Nuclear Power Plants

The Committee met with representatives of the NRC staff and Electric Power Research Institute (EPRI) to discuss the final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities." The ACRS Subcommittee on Fire Protection also reviewed this matter during its meeting on May 4, 2005. The Office of Nuclear Regulatory Research (RES) and EPRI have completed a cooperative program to consolidate the fire PRA research and development activities, conducted over the past few years, into a single state-of-the-art methodology for fire PRA. The results, documented in NUREG/CR-6850, provide a structured framework for the overall analysis as well as specific recommended practices to address key aspects of the analysis. This work was conducted under the terms of an EPRI/RES Memorandum of Understanding and an accompanying fire research addendum.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations, dated June 10, 2005, recommending that NUREG/CR-6850 be issued. The Committee also recommended that full-scope pilot fire PRAs based on the procedures and methods in NUREG/CR-6850 be completed, and the insights provided by these applications be used to enhance the methodology. The Committee further recommended that efforts be continued to further identify, quantify, and document remaining fire PRA uncertainties.

4. Draft Commission Paper on "Risk-Informed Alternatives to the Single Failure Criterion"

The Committee met with representatives of the NRC staff to review the draft Commission Paper, "Risk-Informed Alternatives to the Single Failure Criterion." The staff developed the alternatives in response to a Staff Requirements Memorandum (SRM), dated March 31, 2003. The staff described the three proposed alternatives included in the draft Commission paper and stated that it is premature to select any particular alternative at this time. The RES staff provided a brief summary of the comments received from the Office of Nuclear Reactor Regulation (NRR) and the efforts being taken to address these comments.

Committee Action

The Committee issued a report to the NRC Chairman on June 10, 2005, concluding that the staff has conducted a useful review of the role of the single failure criterion in the current regulatory system, defined desirable attributes of risk-informed alternatives, and developed some potential alternatives to the single failure criterion. The Committee recommended that additional input from stakeholders be sought to determine if there is sufficient benefit to justify the resources that will be required to proceed with the development of a risk-informed alternative.

5. Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"

The Committee met with representatives of the NRC staff to review the draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," which endorses, with certain exceptions, the Nuclear Energy Institute (NEI) document NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c))." The ACRS Subcommittee on Fire Protection also met with representatives of the NRC staff and NEI on May 17, 2005 to review this matter. 10 CFR 50.48(c) allows licensees to voluntarily adopt and maintain a fire protection program that meets the requirements of National Fire Protection Association Standard 805 (NFPA 805). NEI has worked with representatives of the industry and the NRC staff to develop implementing guidance, NEI 04-02, for the specific provisions of NFPA 805 and 10 CFR 50.48(c). The Regulatory Guide endorses NEI 04-02, with certain exceptions, and offers guidance to licensees in meeting the Commission's requirements.

Committee Action

The Committee issued a report to the NRC Executive Director for Operations, dated June 14, 2005, recommending that the Regulatory Guide not be issued in its present form. The Committee recommended that the acceptability of changes in the fire protection program, that is based on NFPA 805, be determined using methods consistent with Regulatory Guide 1.174. The Committee concluded that NEI 04-02 contains many statements that are inconsistent with the Commission's policy of promoting the use of PRA methods. The Committee recommended that the staff make it clear in the regulatory guide that it does not endorse such statements. The Committee further recommended that the staff ensure that the parts of NEI 04-02 that it endorses use correct methodology and language. The Committee would like to review the revised Regulatory Guide.

6. Status Reports on the Quality Assessment of Selected Research Projects

RES has been charged by the Executive Director for Operations to establish a process to evaluate the effectiveness (quality) and utility of its research programs. This evaluation is mandated by the Government Performance and Results Act. The Committee is assisting RES in assessing the effectiveness and utility of the NRC research programs. The Committee previously selected four specific projects from the list of nine candidate projects provided by RES for quality review for FY 2005. The Committee discussed the status of the quality assessment of two of the selected research projects; (1) thermal-hydraulic test program at the

Penn State University, and (2) containment capacity study by the Sandia National Laboratory (SNL). The Committee concluded that additional documentation on the test results were needed. RES has agreed to provide such documentation at the end of June 2005. Also, since relevant documentation on the SNL containment capacity study will not be made available to the Committee in a timely manner, the Committee and RES have agreed not to evaluate the quality of this project at this time.

Committee Action

The Committee plans to continue its discussion of this matter during future meetings.

7. Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application

The Committee heard presentations by and held discussions with representatives of System Energy Resources, Inc. (SERI), the applicant, and the NRC staff regarding the Grand Gulf early site permit (ESP) application and the associated draft safety evaluation report (DSER).

On October 16, 2003, SERI submitted an ESP application for the Grand Gulf site. The Grand Gulf site is in Claiborne County near Port Gibson, Mississippi, approximately 25 miles south of Vicksburg, Mississippi, and is adjacent to the existing nuclear power reactor operated by Entergy Operations, Inc. The ESP site identified in the application is collocated with the Grand Gulf Nuclear Station, Unit 1. In its application, SERI seeks approval of an ESP that could support a future application to construct and operate additional nuclear unit(s) at the Grand Gulf site, with a total nuclear generating capacity of up to 8,600 megawatts thermal (Mwt), with maximum 4,300 Mwt per unit.

The DSER summarizes the results of the staff's technical evaluation of the suitability of the proposed site for a nuclear power plant(s) falling within the plant parameter envelope that SERI specified in its application.

Committee Action

The Committee issued an interim letter to the NRC Executive Director for Operations regarding this matter dated June 14, 2005. The Committee stated that the NRC staff is preparing a quality safety evaluation report. However, the DSER needs to be augmented with a more complete exposition on threats posed by transportation accidents on the river adjacent to the proposed site.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- There were no EDO responses made available to the Committee for reconciliation.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from May 4, 2005 through June 1, 2005, the following Subcommittee meetings were held:

- Early Site Permits - May 16, 2005

  The Subcommittee met with representatives of the NRC staff and System Energy Resources, Inc., to review the draft Safety Evaluation Report related to early site permit application for the Grand Gulf site.

- Fire Protection - May 17, 2005

  The Subcommittee met with representatives of the NRC staff and NEI to review the draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants."

- Plant License Renewal - May 31, 2005

  The Subcommittee met with representatives of the NRC staff and the Nuclear Management Company, LLC to review the license renewal application for Point Beach Nuclear Plant Units 1 and 2 and the related SER with open items prepared by the staff.

- Planning and Procedures - May 31, 2005

  The Subcommittee discussed proposed ACRS activities, practices, and procedures for conducting Committee business and organizational and personnel matters relating to ACRS and its staff.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee plans to hold additional discussions with the staff on issues related to the licensing of new plants such as the appropriate measures of societal risk, frequency-consequence expressions of acceptance values, and dealing with the prevention goal for multiple modules on a site.

- The Committee plans to review the revised Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants."

- The Committee plans to review the proposed Bulletin, "Emergency Preparedness and Response Actions for Security-Based Events," after the staff has reconciled the industry responses.

- During its review of the final SER, related to the license renewal application for the Point Beach Nuclear Plant Units 1 and 2, the Committee would like to hear about additional actions, such as an expanded review of the license renewal commitments and a focused review of the effectiveness of the corrective action program, before the Point Beach Nuclear Plant enters the period of extended operation.

- The Committee continues to question the defensibility of the methods used by the staff and the applicant to prognosticate the weather at the Grand Gulf site over the next 65 years based just on historical frequencies of severe weather events. At the very minimum, the staff should review current literature on possible changes in weather in the upper Gulf of Mexico to be confident that the methods used for weather predictions are defensible.

PROPOSED SCHEDULE FOR THE 524th ACRS MEETING

The Committee agreed to consider the following topics during the 524th ACRS meeting, to be held on July 6-8, 2005:

- Final Review of the License Renewal Application for Donald C. Cook Nuclear Plant, Units 1 and 2
- Final Safety Evaluation Report Related to North Anna Early Site Permit Application
- Draft Final Regulatory Guide, DG-1137, "Guidelines for Lightning Protection for Nuclear Power Plants"
- Draft Final Revision 2 to Regulatory Guide 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants"
- Status Report/Interim Results of the Quality Assessment of Selected NRC Search Projects

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

CERTIFIED

Date Issued: 7/22/05
Date Certified: 8/1/05

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V. Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion (Open)

VI. Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application (Open)

VII. Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (Open)

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A. Reconciliation of ACRS Comments and Recommendations

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REPORTS:

The following reports to Nils J. Diaz, Chairman, NRC, from Graham B. Wallis, Chairman, ACRS:

- Draft Commission Paper on "Risk-Informed Alternatives to the Single Failure Criterion," dated June 10, 2005

LETTERS:

The following letters to William D. Travers, Executive Director for Operations, NRC, from Graham B. Wallis, Chairman, ACRS:

- Interim Report on the Safety Aspects of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2, dated June 9, 2005
- Draft Final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," dated June 10, 2005
- Interim Letter: Draft Safety Evaluation Report on Grand Gulf Early Site Permit Application, dated June 14, 2005
- Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," dated June 10, 2005

MEMORANDA:

The following memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Proposed Bulletin Entitled, "Emergency Preparedness and Response Actions for Security-Based Events," dated June 3, 2005
- Draft Final Regulatory Guides on ASME Code Cases, dated June 7, 2005

APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
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# MINUTES OF THE 523rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
June 1-3, 2005
ROCKVILLE, MARYLAND

The 523rd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on June 1-3, 2005. Notice of this meeting was published in the *Federal Register* on May 18, 2005 (70 FR 28586) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting is available in the NRC's Public Document Room at One White Flint North, Room 1F-19, 11555 Rockville Pike, Rockville, Maryland. Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc. 1323 Rhode Island Avenue, NW, Washington, DC 20005. Transcripts are also available at no cost to download from, or review on, the Internet at http://www.nrc.gov/ACRS/ACNW.

ATTENDEES

ACRS Members: ACRS Members: Dr. Graham B. Wallis (Chairman), Dr. William J. Shack (Vice Chairman), Mr. John D. Sieber, (Member-at-Large), Dr. George E. Apostolakis, Dr. Mario V. Bonaca, Dr. Richard S. Denning, Dr. Thomas S. Kress, Dr. Dana A. Powers, Dr. Victor H. Ransom, and Mr. Stephen L. Rosen. For a list of other attendees, see Appendix III.

## I. Chairman's Report (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

Dr. Graham B. Wallis, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee.

## II. Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2

[Note: Mr. Cayetano Santos was the Designated Federal Official for this portion of the meeting.]

The Committee met with NRC staff and representatives of the Nuclear Management Company, LLC (NMC) to perform an interim review of the license renewal application of the Point Beach Nuclear Plant, Units 1 and 2 and the related Safety Evaluation Report (SER) with open items. The operating licenses for Units 1 and 2 expire on October 5, 2010, and March 8, 2013,

respectively. The applicant has requested approval for continued operation of each unit for a period of 20 years beyond the current license expiration dates.

Mr. Bonaca, Chairman of the Plant License Renewal Subcommittee, stated that Point Beach Unit 2 is projected to exceed the pressurized thermal shock screening criterion in 2017 and the applicant has committed to implement a flux reduction program in accordance with 10 CFR 50.61. The Committee recognizes that current operating performance is not a consideration in the license renewal rule, but is concerned that aspects of current performance may affect the development and effective implementation of license renewal programs.

Mr. Cooper, NMC Senior Vice President, presented the Site Excellence Plan for sustaining improvement at Point Beach Nuclear Plant. Closure of the Confirmatory Action Letter (CAL) is a critical milestone and 134 of 143 items to address the CAL have been completed. Mr. Cooper stated that most items will be completed by the end of the year, but some engineering-related calculations will be completed in 2006.

Mr. Louden, Region III Branch Chief, described the "Red" findings issued under the Reactor Oversight Process, the CAL issued on April 21, 2004, and the additional inspections performed by the staff. The five areas of concern outlined in the CAL are human performance, engineering design control, the engineering/operations interface, emergency preparedness, and the Corrective Action Program. The applicant has made progress in all these areas but the CAL remains open. The staff also identified substantive cross-cutting issues in human performance and problem identification and resolution. Mr. Louden stated that the applicant's Corrective Action Program is sound, but improvement is still needed in some aspects of its implementation.

Committee Action

The Committee issued a letter to the EDO, dated June 9, 2005, describing its concerns with the applicant's corrective action program and human performance that may affect the development and effective implementation of license renewal programs. The Committee would like to hear what additional actions the staff plans to take to increase confidence that the requirements of the license renewal rule have been met and there is reasonable assurance that aging degradation can be adequately managed.

III. Draft Commission Paper on Policy Issues Related to New Plant Licensing

[Note: Mr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the draft Commission paper on policy issues related to new plant licensing. The Committee was briefed on two policy issues: the minimum level of safety that new plants need to meet to achieve enhanced safety; and how to account for the risk from multiple reactors at a single site.

The staff has defined four options for the level of safety issue. The staff recommends the option that would establish the Quantitative Health Objectives (QHOs) as the minimum level of safety and the regulatory framework will be designed to meet this level of safety. Such an approach goes beyond current practice in that compliance with the QHOs would be mandatory.

For the integrated risk issue, the staff has defined three options. The staff recommends the option that the integrated risk from all new units at the site satisfy the Commission's QHOs.

Committee Action

The Committee plans to continue its debate on these two policy issues during its July 6-8, 2005 meeting.

IV. Fire Risk Requantification and Probabilistic Risk Assessment (PRA) Methodology for Nuclear Power Plants

[Note: Mr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and Electric Power Research Institute (EPRI) to discuss the final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities." The ACRS Subcommittee on Fire Protection also reviewed this matter during its meeting on May 4, 2005. The Office of Nuclear Regulatory Research (RES) and EPRI have completed a cooperative program to consolidate the fire PRA research and development activities, conducted over the past few years, into a single state-of-the-art methodology for fire PRA. The results, documented in NUREG/CR-6850, provide a structured framework for the overall analysis as well as specific recommended practices to address key aspects of the analysis. This work was conducted under the terms of an EPRI/RES Memorandum of Understanding and an accompanying fire research addendum.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations, dated June 10, 2005, recommending that NUREG/CR-6850 be issued. The Committee also recommended that full-scope pilot fire PRAs based on the procedures and methods in NUREG/CR-6850 be completed, and the insights provided by these applications be used to enhance the methodology. The Committee further recommended that efforts be continued to further identify, quantify, and document remaining fire PRA uncertainties.

V. Draft Commission Paper on "Risk-Informed Alternatives to the Single Failure Criterion"

[Note: Mr. Michael Snodderly was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff to review the draft Commission Paper, "Risk-Informed Alternatives to the Single Failure Criterion." The staff developed the alternatives in response to a Staff Requirements Memorandum (SRM), dated March 31, 2003. The staff described the three proposed alternatives included in the draft Commission paper and stated that it is premature to select any particular alternative at this time. The RES staff provided a brief summary of the comments received from the Office of Nuclear Reactor Regulation (NRR) and the efforts being taken to address these comments.

Committee Action

The Committee issued a report to the NRC Chairman on June 10, 2005, concluding that the staff has conducted a useful review of the role of the single failure criterion in the current regulatory system, defined desirable attributes of risk-informed alternatives, and developed some potential alternatives to the single failure criterion. The Committee recommended that additional input from stakeholders be sought to determine if there is sufficient benefit to justify the resources that will be required to proceed with the development of a risk-informed alternative.

VI. Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application

[Note: Mr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

The Committee heard presentations by and held discussions with representatives of System Energy Resources, Inc. (SERI), the applicant, and the NRC staff regarding the Grand Gulf early site permit (ESP) application and the associated draft safety evaluation report (DSER).

On October 16, 2003, SERI submitted an ESP application for the Grand Gulf site. The Grand Gulf site is in Claiborne County near Port Gibson, Mississippi, approximately 25 miles south of Vicksburg, Mississippi, and is adjacent to the existing nuclear power reactor operated by Entergy Operations, Inc. The ESP site identified in the application is collocated with the Grand Gulf Nuclear Station, Unit 1. In its application, SERI seeks approval of an ESP that could support a future application to construct and operate additional nuclear unit(s) at the Grand Gulf site, with a total nuclear generating capacity of up to 8,600 megawatts thermal (Mwt), with maximum 4,300 Mwt per unit.

The DSER summarizes the results of the staff's technical evaluation of the suitability of the proposed site for a nuclear power plant(s) falling within the plant parameter envelope that SERI specified in its application.

Committee Action

The Committee issued an interim letter to the NRC Executive Director for Operations regarding this matter dated June 14, 2005. The Committee stated that the NRC staff is preparing a quality safety evaluation report. However, the DSER needs to be augmented with a more complete exposition on threats posed by transportation accidents on the river adjacent to the proposed site.

VII. Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"

[Note: Mr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff to review the draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," which endorses, with certain exceptions, the Nuclear Energy Institute (NEI) document NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c))." The ACRS Subcommittee on Fire Protection also met with representatives of the NRC staff and NEI on May 17, 2005 to review this matter. 10 CFR 50.48(c) allows licensees to voluntarily adopt and maintain a fire protection program that meets the requirements of National Fire Protection Association Standard 805 (NFPA 805). NEI has worked with representatives of the industry and the NRC staff to develop implementing guidance, NEI 04-02, for the specific provisions of NFPA 805 and 10 CFR 50.48(c). The Regulatory Guide endorses NEI 04-02, with certain exceptions, and offers guidance to licensees in meeting the Commission's requirements.

Committee Action

The Committee issued a report to the NRC Executive Director for Operations, dated June 14, 2005, recommending that the Regulatory Guide not be issued in its present form. The Committee recommended that the acceptability of changes in the fire protection program, that is based on NFPA 805, be determined using methods consistent with Regulatory Guide 1.174. The Committee concluded that NEI 04-02 contains many statements that are inconsistent with the Commission's policy of promoting the use of PRA methods. The Committee recommended that the staff make it clear in the regulatory guide that it does not endorse such statements. The Committee further recommended that the staff ensure that the parts of NEI 04-02 that it endorses use correct methodology and language. The Committee would like to review the revised Regulatory Guide.

VIII. Status Reports on the Quality Assessment of Selected Research Projects

[Note: Mr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

RES has been charged by the Executive Director for Operations to establish a process to evaluate the effectiveness (quality) and utility of its research programs. This evaluation is mandated by the Government Performance and Results Act. The Committee is assisting RES in assessing the effectiveness and utility of the NRC research programs. The Committee previously selected four specific projects from the list of nine candidate projects provided by RES for quality review for FY 2005. The Committee discussed the status of the quality assessment of two of the selected research projects; (1) thermal-hydraulic test program at the Penn State University, and (2) containment capacity study by the Sandia National Laboratory (SNL). The Committee concluded that additional documentation on the test results were needed. RES has agreed to provide such documentation at the end of June 2005. Also, since relevant documentation on the SNL containment capacity study will not be made available to the Committee in a timely manner, the Committee and RES have agreed not to evaluate the quality of this project at this time.

Committee Action

The Committee plans to continue its discussion of this matter during future meetings.

IX. Executive Session (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

The Committee discussed the response from the NRC Executive Director for Operations (EDO) to ACRS comments and recommendations included in recent ACRS reports:

- There were no EDO responses made available to the Committee for reconciliation.

B. Report on the Meeting of the Planning and Procedures Subcommittee (Open)

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on May 31, 2005. The following items were discussed:

Review of the Member Assignments and Priorities for ACRS Reports and Letters for the June ACRS meeting

Member assignments and priorities for ACRS reports and letters for the June ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were also discussed.

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through September 2005 were addressed. The objectives were:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action.

Meeting with the EDO, Deputy EDOs, and Program Office Directors

The ACRS is scheduled to meet with the EDO, Deputy EDOs, and Office Directors of NRR, RES, and NMSS between 10:00 a.m. and 12 noon, on September 9, 2005 to discuss items of mutual interest. ACNW members normally participate in this session. The members should identify topics for discussion during this meeting.

Meeting with the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners on Thursday, October 6, 2005, to discuss items of mutual interest. The members should propose topics during the June or July ACRS meeting.

Quadripartite Meeting Status

During its May 2005 meeting, the Committee discussed the comments received from GPR, NSC, and RSK on the dates and proposed topics for the Quadripartite meeting. All Countries agreed to hold this meeting on October 18-20, 2006. Also, they are in support of having the meeting on the east coast of the United States and we are currently planning on holding it in the Washington, DC area. A revised list of topics, which reflects incorporation of the comments received from GPR, NSC, and RSK as well as the comments provided by the members at the May were discussed. A revised list of topics were sent to all Countries.

Letter from ASTM

In a letter dated May 16, 2005, Roger Stoller, Chairman of the ASTM Committee E10 on Nuclear Technology and Applications, requested Dr. Wallis, Chairman of the ACRS, for "support in encouraging increased and consistent participation by the NRC in ASTM's standards development work." ASTM revised E900, the Standard Guide for Predicting Radiation-Induced Transition Temperature Shifts in Reactor Vessel Materials, in parallel with an NRC revision of Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials. He stated that the staff has not consistently participated in ASTM meetings and it appears that the E900 Standard and proposed Revision 3 to Regulatory Guide 1.99 will be different. The letter notes that Federal agencies are required to use standards developed by consensus standards bodies except where that would be inconsistent with applicable law or otherwise impractical.

Self-Assessment of ACRS Performance

The ACRS self-assessment SECY paper went to the Commission on June 30, 2005. The ACRS Executive Director and/or the Deputy Executive Director met with the Commissioners, EDO, and the Program Office Directors and obtained feedback on the ACRS/ACNW performance. Feedback from several external stakeholders were also received. A draft Commission Paper has been prepared and it will be provided to the P&P for comment prior to issuing.

Safeguards and Security Matters

In a recent Staff Requirements Memorandum, the Commission directed the staff to perform two additional selected confirmatory spent fuel pool analyses in the area previously recommended by the ACRS, to provide added assurance that the issue is not as significant, as the staff previously determined.

According to the May 19, 2005 Nuclear News Flashes, the House Appropriations Committee approved an additional $21 million for the NRC to perform technical analyses of spent fuel storage risks and respond to other security-related recommendations of the National Academy of Sciences.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 524th ACRS Meeting, July 6-8, 2005.

The 523rd ACRS meeting was adjourned at 6:47 p.m. on June 3, 2005.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 22, 2005

MEMORANDUM TO: ACRS Members

FROM: Noble S. Green, Jr.
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 523rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - June 1-3, 2005

Enclosed are the proposed minutes of the 523rd meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate, which will be distributed within six (6) working days from the date of this memorandum.

Attachment:
As stated

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 1, 2005

MEMORANDUM TO: Noble S. Green, Jr., Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Graham B. Wallis
ACRS Chairman

SUBJECT: CERTIFIED MINUTES OF THE 523rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), June 1-3, 2005

I certify that based on my review of the minutes from the 523rd ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

Director, Division of Nuclear Materials Safety, NRC Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, a request for termination of License No. 50–23220–02, using NRC Form 314.

The Regional Administrator, Region IV, may relax or rescind, in writing, any of the above conditions upon a showing by the Licensee of good cause.

In accordance with 10 CFR 2.202, any person adversely affected by this Confirmatory Order, other than the Licensee, may request a hearing on this Order within 20 days of its issuance. Where good cause is shown, consideration will be given to extending the time to request a hearing. A request for extension of time must be made in writing to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. Any request for a hearing shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, ATTN: Rulemakings and Adjudications Staff, Washington, DC 20555. Copies also shall be sent to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, to the Assistant General Counsel for Materials Litigation and Enforcement at the same address, to the Regional Administrator, NRC Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, TX 76011, and to the Licensee. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that answers and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov* and also to the Office of the General Counsel either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov*. If a person other than the licensee requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.309(d)(1) and (f).

If a hearing is requested by a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing shall be whether this Confirmatory Order should be sustained. In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final 20 days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received. An answer or a request for hearing shall not stay the immediate effectiveness of this order.

For the Nuclear Regulatory Commission.

Dated this 9th day of May, 2005.

**Frank J. Congel,**

*Director, Office of Enforcement.*

[FR Doc. E5–2490 Filed 5–17–05; 8:45 am]

BILLING CODE 7590–01–P

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## NUCLEAR REGULATORY COMMISSION

### Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on June 1–3, 2005, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Wednesday, November 24, 2004 (69 FR 68412).

**Wednesday, June 1, 2005, Conference Room T–2b3, Two White Flint North, Rockville, Maryland**

*8:30 a.m.—8:35 a.m.: Opening Remarks by the ACRS Chairman* (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

*8:35 a.m.—9:45 a.m.: Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2* (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Nuclear Management Company, LLC regarding the license renewal application for the Point Beach Nuclear Plant, Units 1 and 2 and the associated draft Safety Evaluation Report prepared by the NRC staff, as well as the progress being made by the NRC staff and the applicant in resolving the issue of potential common-mode failure of the auxiliary feedwater pumps due to operator actions specified in the plant procedures, and related issues.

*10 a.m.—11:30 a.m.: Draft Commission Paper on Policy Issues Related to New Plant Licensing* (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft Commission paper on policy issues (integrated risk and level of safety) related to new plant licensing.

*12:30 p.m.—2 p.m.: Fire Risk Requantification and Probabilistic Risk Analysis (PRA) Methodology for Nuclear Power Plants* (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Electric Power Research Institute (EPRI) regarding draft final NUREG/CR–6850, "EPRI/NRC–RES Fire PRA Methodology for Nuclear Power Facilities," and related matters.

*2:15 p.m.—4:15 p.m.: Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion* (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft Commission Paper on the proposed risk-informed and performance-based alternatives to the existing single failure criterion.

*4:30 p.m.—7 p.m.: Preparation of ACRS Reports* (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting as well as a proposed report responding to the Commission request in the April 26, 2005 Staff Requirements Memorandum regarding the ACRS assessment of the quality of the NRC research projects.

**Thursday, June 2, 2005, Conference Room T–2b3, Two White Flint North, Rockville, Maryland**

*8:30 a.m.—8:35 a.m.: Opening Remarks by the ACRS Chairman* (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

*8:35 a.m.—10 a.m.: Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application* (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and System Energy Resources Inc. regarding the NRC staff's draft Safety Evaluation Report related to the Grand Gulf Early Site Permit Application.

*10:15 a.m.—11:45 a.m.: Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants"* (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding the draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," which endorses, with certain exceptions, NEI document, NEI 04–02, "Guidance for Implementing a

Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)," and the NRC staff's resolution of public comments.

*12:45 p.m.—1:45 p.m.: Status Report on the Quality Assessment of Selected Research Projects* (Open)—The Committee will hear a report by the Chairmen of the ACRS Panels regarding the status of the assessment of the quality of the thermal-hydraulic test program at the Penn State University and the containment capacity study being performed by the Sandia National Laboratories.

*1:45 p.m.—2:30 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee* (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

*2:30 p.m.—2:45 p.m.: Reconciliation of ACRS Comments and Recommendations* (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

*3 p.m.—7 p.m.: Preparation of ACRS Reports* (Open)—The Committee will discuss proposed ACRS reports.

**Friday, June 3, 2005, Conference Room T-2b3, Two White Flint North, Rockville, Maryland**

*8:30 a.m.—5 p.m.: Preparation of ACRS Reports* (Open)—The Committee will continue its discussion of proposed ACRS reports.

*5 p.m.—5:30 p.m.: Miscellaneous* (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 5, 2004 (69 FR 59620). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Cognizant ACRS staff (301–415–7364), between 7:30 a.m. and 4:15 p.m., e.t.

ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., e.t., at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: May 12, 2005.

**Annette Vietti-Cook,**

*Secretary of the Commission.*

[FR Doc. E5–2489 Filed 5–17–05; 8:45 am]

BILLING CODE 7590–01–P

## SECURITIES AND EXCHANGE COMMISSION

**[Release No. 34–51685; File No. SR–Amex–2005–050]**

## Self-Regulatory Organizations; American Stock Exchange LLC; Notice of Filing and Immediate Effectiveness of a Proposed Rule Change Relating to the Options Marketing Fee for Options on SPDRs

May 11, 2005.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 ("Act"),[1] and Rule 19b–4 thereunder,[2] notice is hereby given that on April 28, 2005, the American Stock Exchange LLC ("Amex" or "Exchange") filed with the Securities and Exchange Commission ("Commission") the proposed rule change as described in Items I, II and III below, which Items have been prepared by the Amex. The Amex has designated this proposal as one establishing or changing a due, fee, or other charge imposed by the Amex under Section 19(b)(3)(A)(ii) of the Act,[3] and Rule 19b–4(f)(2) thereunder,[4] which renders the proposal effective upon filing with the Commission. The Commission is publishing this notice to solicit comments on the proposed rule change from interested persons.

### I. Self-Regulatory Organization's Statement of the Terms of Substance of the Proposed Rule Change

The Exchange proposes to amend its marketing fee imposed on certain transactions of specialists and registered options traders ("ROTs") in connection with options on Standard & Poor's Depositary Receipts ("SPDRs"). The fee would be imposed at the rate of $1.00 per contract, instead of the rate of $0.40 per contract. The text of the proposed rule change is available on the Amex's Web site (*http://www.amex.com*), at the Amex's Office of the Secretary, and at the Commission's Public Reference Room.

### II. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

In its filing with the Commission, the Exchange included statements concerning the purpose of and basis for its proposal and discussed any comments it had received regarding the proposal. The text of these statements may be examined at the places specified

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.

[3] 15 U.S.C. 78s(b)(3)(A)(ii).

[4] 17 CFR 240.19b–4(f)(2).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

**May 11, 2005**

**SCHEDULE AND OUTLINE FOR DISCUSSION**
**523rd ACRS MEETING**
**JUNE 1-3, 2005**

**WEDNESDAY, JUNE 1, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
   1.1) Opening statement
   1.2) Items of current interest

2) 8:~~35~~ 45 - 9:45 A.M. Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2 (Open) (MVB/CS)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Management Company, LLC regarding the license renewal application for the Point Beach Nuclear Plant, Units 1 and 2 and the associated draft Safety Evaluation Report prepared by the NRC staff, as well as the progress being made by the NRC staff and the applicant in resolving the issue of potential common-mode failure of the auxiliary feedwater pumps due to operator actions specified in the plant procedures, and related issues.

**9:~~45~~ 50 - 10:~~00~~ 06 A.M. ***BREAK*****

3) 10:~~00~~ 06 - 11:~~30~~ 46 A.M. Draft Commission Paper on Policy Issues Related to New Plant Licensing (Open) (TSK/MME)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding the draft Commission paper on policy issues (integrated risk and level of safety) related to new plant licensing.

**11:~~30~~ 46 - 12:~~30~~ 45 P.M. ***LUNCH*****

4) 12:~~30~~ 45 - 2:~~00~~ 04 P.M. Fire Risk Requantification and Probabilistic Risk Analysis (PRA) Methodology for Nuclear Power Plants (Open) (SLR/HPN)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff and Electric Power Research Institute (EPRI) regarding the draft final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," and related matters.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

**2:00 - 2:15 P.M.** ***BREAK***

5) 2:15 - 4:15 P.M. Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion (Open) (WJS/MRS)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the draft Commission Paper on the proposed risk-informed and performance-based alternatives to the existing single failure criterion.

Do not need court Reporter for the remainder of the meeting, per Dr. Wallis.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

**4:15 - 4:30 P.M.** ***BREAK***

6) 4:30 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2 (MVB/CS)

6.2) Draft Commission Paper on Policy Issues Related to New Plant Licensing (TSK/MME)

6.3) Draft Final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" (SLR/HPN)

6.4) Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion (WJS/MRS)

6.5) Response to the April 26, 2005 Staff Requirements Memorandum Regarding the ACRS Assessment of the Quality of the NRC Research Projects (Tentative) (DAP/HPN/SD)

**THURSDAY, JUNE 2, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**

7) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)

8) 8:35 - 10:00 A.M. Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application (Open) (DAP/MME)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff and System Energy Resources Inc. regarding the NRC staff's draft Safety Evaluation Report related to the Grand Gulf Early Site Permit Application.

**10:00 - 10:15 A.M.** ***BREAK***

9) 10:~~15~~ 37 - ~~11:45~~ 12:00 A.M. Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (Open) (GEA/HPN)

9.1) Remarks by the Cognizant ACRS Member

9.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding the draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," which endorses, with certain exceptions, NEI document, NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)," and the NRC staff's resolution of public comments.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

~~**11:45 - 12:45 P.M.**~~ 12:00 - 1:00 P.M. *****LUNCH*****

10) ~~12:45~~ 1:00 - 1:45 P.M. Status Reports on the Quality Assessment of Selected Research Projects (Open) (DAP/GBW/WJS/RC/EAT)

10.1) Report by the Chairman of the ACRS Panel regarding the status of the assessment of the quality of the thermal-hydraulic test program at the Penn State University.

10.2) Report by the Chairman of the ACRS Panel on the assessment of the quality of the containment capacity study being performed by the Sandia National Laboratories.

11) 1:45 - 2:30 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

11.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

11.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

12) 2:30 - 2:45 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)

Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

~~**2:45 - 3:00 P.M.**~~ 3:05 30 *****BREAK*****

13) 3:~~00~~ 30 - ~~7:00~~ 6:40 P.M. P.M.

4:58-5:15 p.m. Break

Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

13.1) Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2 (MVB/CS)
13.2) Draft Commission Paper on Policy Issues Related to New Plant Licensing (TSK/MME)
13.3) Draft Final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" (SLR/HPN)
13.4) Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion (WJS/MRS)
13.5) Draft Safety Evaluation Report on the Grand Gulf Early Site Permit Application (DAP/MME)
13.6) Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (GEA/HPN)
13.7) Response to the April 26, 2005 Staff Requirements Memorandum Regarding the ACRS Assessment of the Quality of the NRC Research Projects (Tentative) (DAP/HPN/SD)

**FRIDAY, JUNE 3, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**

14) 8:30 - 5:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 13.

15) 5:00 - ~~5:30~~ 6:47 P.M. Miscellaneous (Open) (GBW/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

**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.**
- **Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.**

APPENDIX III: MEETING ATTENDEES

523rd ACRS MEETING
June 1-3, 2005

NRC STAFF (6/1/05)

Y. Diaz, NRR
K. Alm-Lytz, NRR
P. Longheed, RIII
K. Weaver, NRR
V. Rodriguez, NRR
P. Louden, RIII
M. Morgan, NRR
L. Miller, NRR
D. Merzke, NRR
K. Cheng, NRR
S. Lee, NRR
J. Rowley, NRR
J. Medoff, NRR
T. Koslty, NRR
N. Patel, NRR
R. Subbaratnan, NRR
S.K. Nibin, NRR
K. Cozer, NRR
R. McIntyre, NRR
T. Le, NRR
H. Chernoff, NRR
S. Ray, NRR
D. Matthews, NRR
N. Ray, NRR
M. Drouin, RES
N. Iqbal, NRR
M. Stutzke, NRR
G. Galletti, NRR
D. Lew, RES
J. Levin, RES
R. Gallucci, NRR
P. Lain, NRR
S.M. Wong, NRR
M. H. Salley, RES
A. Klein, NRR
J. Mitchell, RES
G. Parry, NRR
D. Frankin, NRR
D. Harrison, NRR
S. A. Laur, NRR
S. Dinsmore, NRR
H. Hamzehee, RES
J. C. Lane, RES
A. Salomon, RES
J. Lyons, NRR
M. Tschiltz, NRR
A. Buslik, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

D. Turner, NMC-Palisades
P. Kollath-Rorhno, RPI
M. Heath, PLN
J. G. Thorgersen, NMC-PBNA
J. Weil, McGraw-Hill
T. King, Self
A. Levin, FANP
S. G. Sterrett, Self (Duke University)
F. Jaglan, SAIC
S. Nowlen, Sandia Lab's
B. Najafi, SAIC/EPRI
B. Youngblood, ISL
S. Newberry, ISL
L. Cambell, BNL
G. Martinez-Guridi, BNL
S. Traiforos, LINK
W. D. Crouch, TVA
J. Thomas, First Energy
S. Dert, First Energy
K. Brune, TVA
R. Jennings, TVA
S. Sohellin, NMC-PBNP
D. Arp, TVA
R. Jansen, TVA
D. Cooper, NMC
J. Schweitzer, NMC
B. Herrman, NMC
R. Vincent, NMC
R. Grumbir, AEP
D. Johnson, NMC
B. Fromm, NMC
T. Mielke, NMC
J. Knorl, NMC
M. Ortmayer, NMC

APPENDIX III: MEETING ATTENDEES (Cont'd)

523rd ACRS MEETING
June 1-3, 2005

NRC STAFF (6/2/05)
S.M. Wong, NRR
R. Dipert, NRR
S. Black, NRR
P. Lain, NRR
N. Cholesh, RES
P. O'Reilly, RES
R. McIntyre, NRR
M. H. Sally, RES
V. Rodriguez, NRR
S. Lee, NRR
F. Eltawila, RES

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
M. Bourgeors, Entergy
G. Zinke, Entergy
P. Campbell, Morgan Lewis
B. Hoffman, Public Citizen
J. Weil, McGraw-Hill
B. Najafi, SAIC/EPRI
S. Traniforos, LINK

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 8, 2005

# SCHEDULE AND OUTLINE FOR DISCUSSION
# 524th ACRS MEETING
# JULY 6-8,2005

**WEDNESDAY, JULY 6, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)
   - 1.1) Opening statement
   - 1.2) Items of current interest

2) 8:35 - 10:00 A.M. Final Review of the License Renewal Application for Donald C. Cook Nuclear Plant, Units 1 and 2 (Open) (MVB/CS/JGL)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the Indiana Michigan Power Company and the NRC staff regarding the license renewal application for Donald C. Cook Nuclear Plant, Units 1 and 2 and the associated final Safety Evaluation Report prepared by the NRC staff.

**10:00 - 10:15 A.M. ***BREAK*****

3) 10:15 - 12:15 P.M. Final Safety Evaluation Report Related to North Anna Early Site Permit Application (Open) (DAP/MME)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the Dominion North Anna, LLC and the NRC staff regarding the NRC staff's Final Safety Evaluation report related to the North Anna Early Site Permit Application.

   Members of the public may provide their views, as appropriate.

**12:15 - 1:45 P.M. ***LUNCH*****

4) 1:45 - 3:15 P.M. Draft Final Regulatory Guide, DG-1137, "Guidelines for Lightning Protection for Nuclear Power Plants" (Open) (JDS/JGL/MRS)
   - 4.1) Remarks by the Subcommittee Chairman
   - 4.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revision to Regulatory Guide, DG-1137, and the NRC staff's resolution of public comments.

   Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

**3:15 - 3:30 P.M. ***BREAK*****

5) 3:30- 5:00 P.M. Draft Final Revision 2 to Regulatory Guide 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants" (Open) (GEA/EAT)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revision 2 to Regulatory Guide 1.152, and the NRC staff's resolution of public comments.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

**5:00 - 5:15 P.M.** ***BREAK***

6) 5:15 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Final Review of the License Renewal Application for D. C. Cook Nuclear Plant, Units 1 and 2 (MVB/CS/JGL)

6.2) North Anna Early Site Permit Application (DAP/MME)

6.3) Draft Final Regulatory Guide, DG-1137, "Guidelines for Lightning Protection for Nuclear Power Plants" (JDS/JGL/MRS)

6.4) Draft Final Revision 2 to Regulatory Guide 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants" (GEA/EAT)

6.5) Response to the April 26, 2005 Staff Requirements Memorandum Regarding the ACRS Assessment of the Quality of the NRC Research Projects (Tentative) (DAP/HPN/SD)

6.6) Draft Commission Paper on Policy Issues Related to New Plant Licensing (TSK/MME)

**THURSDAY, JULY 7, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**

7) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GBW/JTL/SD)

8) 8:35 - 9:30 A.M. Subcommittee Reports (Open) (GEA/EAT/JGL)

8.1) Report by the Chairman of the ACRS Subcommittee on Digital Instrumentation and Control (I&C) Systems regarding the digital I&C research plan and other related matters that were discussed at the June 14-15, 2005 Subcommittee meeting.

8.2) Report by the Chairman of the Joint ACRS Subcommittee on Reliability and Probabilistic Risk Assessment and on Plant Operations regarding the Risk-Management Technical Specifications and related matters that were discussed at the June 15, 2005 Subcommittee meeting.

9) 9:30 - 10:30 A.M. Status Report/Interim Results of the Quality Assessment of Selected NRC Research Projects (Open) (DAP/GEA/JDS/EAT/CS)

9.1) Report by the Chairman of the ACRS Panel regarding the status/interim results of the quality assessment of the NRC research project on Standardized Plant Analysis Risk (SPAR) models.

9.2) Report by the Chairman of the ACRS Panel regarding the status/interim results of the assessment of the quality of the Steam Generator Tube Integrity Program at the Argonne National Laboratory.

**10:30 - 10:45 A.M.** ***BREAK***

10) 10:45 - 11:45 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GBW/JTL/SD)

10.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

10.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

11) 11:45 - 12:00 Noon Reconciliation of ACRS Comments and Recommendations (Open) (GBW, et al./SD, et al.)

Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

**12:00 - 1:30 P.M.** ***LUNCH***

12) 1:30 - 4:30 P.M. Preparation of ACRS Reports (Open)

**(3:00-3:15 P.M. BREAK)** Discussion of proposed ACRS reports on:

12.1) Final Review of the License Renewal Application for D. C. Cook Nuclear Plant, Units 1 and 2 (MVB/CS/JGL)

12.2) North Anna Early Site Permit Application (DAP/MME)

12.3) Draft Final Regulatory Guide, DG-1137, "Guidelines for Lightning Protection for Nuclear Power Plants" (JDS/JGL/MRS)

12.4) Draft Final Revision 2 to Regulatory Guide 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants" (GEA/EAT)

12.5) Response to the April 26, 2005 Staff Requirements Memorandum Regarding the ACRS Assessment of the Quality of the NRC Research Projects (Tentative) (DAP/HPN/SD)

12.6) Draft Commission Paper on Policy Issues Related to New Plant Licensing (TSK/MME)

**4:30 - 4:45 P.M.** ***BREAK***

13) 4:45 - 6:45 P.M. Safeguards and Security Matters, Room T-8E8 (Closed) (MVB/RPS/EAT)

13.1) Remarks by the Subcommittee Chairman

13.2) Briefing by and discussions with representatives of the NRC staff regarding safeguards and security matters.

**[Note: This session will be closed to protect information classified as national security information and safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).]**

**FRIDAY, JULY 8, 2005, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**

14) 8:30 - 4:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 12.

15) 4:00 - 4:30 P.M. Miscellaneous (Open) (GBW/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

**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.**
- **Thirty-Five (35) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
523rd ACRS MEETING
June 1-3, 2005

[Note: Some documents listed below may have been provided or prepared for Committee use only. These documents must be reviewed prior to release to the public.]

MEETING HANDOUTS

| AGENDA ITEM NO. | DOCUMENTS |
|---|---|
| 1. | Opening Remarks by the ACRS Chairman |
| | 1. Items of Interest, dated June 1-3, 2005 |
| 2. | Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2 |
| | 2. Point Beach Sustaining Improvement presentation by Nuclear Management Company (NMC) [PowerPoint Slides] |
| | 3. Point Beach Discussion of Red Findings presentation by Nuclear Management Company (NMC) [PowerPoint Slides] |
| 3. | Draft Commission Paper on Policy Issues Related to New Plant Licensing |
| | 4. Policy Issues Related to New Plant Licensing presentation by RES [PowerPoint Slides] |
| 4. | Fire Risk Requantification and Probabilistic Risk Analysis (PRA) Methodology for Nuclear Power Plants |
| | 5. EPRI/NRC-RES Fire PRA Methodology/Background Information presentation by RES [PowerPoint Slides] |
| 5. | Draft Commission Paper on Proposed Alternatives to the Existing Single Failure Criterion |
| | 6. Status of Evaluation of a Broader Change to Single Failure Criterion Model Development Program presentation by RES [Viewgraphs] |
| 8. | Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application |
| | 7. Proposed Agenda (for Entergy & NRR Presenters) |
| | 8. Early Site Permit Application for the Grand Gulf Nuclear Station Site presentation by Entergy [Powerpoint Slides] |
| | 9. Safety Review of the Grand Gulf Early Site Permit Application presentation by NRR [Powerpoint Slides] |
| 9. | Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" |
| | 10. Regulatory Guide for NFPA 805 Rule - Fire Protection Section Presentation by NRR [Powerpoint Slides] |

11. Plant Change Evaluation Example for NFPA-805 - Plant Change Evaluation by NRR [Powerpoint Slides]
12. Figure 5-1 - Plant Change Evaluation (Chart) presented by NRR [Viewgraph]

11. Future ACRS Activities/Report of the Planning and Procedures Subcommittee
    13. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - May 31, 2005 [Handout #11]

Appendix V
523rd ACRS Meeting

MEETING NOTEBOOK CONTENTS

Color Code List - 523rd ACRS Meeting
Overtime Schedule

| TAB | DOCUMENTS |
|---|---|

2 Interim Review of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2
1. Table of Contents
2. Proposed Schedule
3. Status Report, dated June 1, 2005

3 Draft Commission Paper on Policy Issues Related to New Plant Licensing
1. Table of Contents
2. Proposed Agenda
3. Status Report
4. Attachments
   A. SRM - SECY-03-0047 dated June 26, 2003
   B. ACRS Letter to Chrm Diaz dated April 22, 2004
   C. ACRS Letter to L. Reyes, EDO dated December 9, 2004
   D. SECY-05-0006 (Information) dated January 7, 2005
   E. Memo to J. Larkins, ACRS from C. Ader, RES dated May 11, 2005

4 Fire Risk Requantification and Probabilistic Risk Assessment (PRA) Methodology for Nuclear Power Plants
1. Table of Contents
2. Proposed Schedule
3. Status Report
4. Attachments
   A. EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Volume 1
   B. EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Volume 2

8 Draft Safety Evaluation Report Related to Grand Gulf Early Site Permit Application
1. Table of Contents
2. Proposed Agenda
3. Status Report
4. Attachments
   A. ACRS Letter, Draft Review Standard, RS-002, dated March 12, 2003
   B. Grand Gulf ESP Application, Section 1.3, "Plant Parameters Envelope

9 Draft Final Regulatory Guide, "Risk-Informed, Performance Based Fire Protection for Existing Light-Water Nuclear Power Plants"

1. Table of Contents
2. Proposed Schedule
3. Status Report
4. Attachments
   - A. Regulatory Guide X.XXX, "Risk-informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants", July 2005
   - B. Nuclear Energy Institute (NEI), "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," NEI 04-02, Revision 0, April 2005
   - C. ACRS Letter from M. Bonaca, Chairman, to N. Diaz, Chairman, NRC, dated December 12, 2003, Subject: Draft Final Rule Revising 10 CFR 50.48, "Fire Protection," to Permit Licensees to Voluntarily Adopt Fire Protection Requirements Contained in National Fire Protection Association Standard 805 (NFPA 805)

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**523rd FULL COMMITTEE MEETING**

**June 1-3, 2005**

**TODAY'S DATE: June 1, 2005**

**NRC STAFF - PLEASE SIGN BELOW**

**PLEASE PRINT (CLEARLY)**

| NAME | NRC ORGANIZATION |
|---|---|
| Yoira Diaz | NRR/DRIP/RLEP |
| Kirsi Alm-Lytz | NRR/IPSB |
| Patricia Longhead | RIII |
| Kathy Weaver | NRR/DRIP/RLEP |
| Veronica Rodriguez | NRR/DRIP/RLEP-A |
| JOHN G. LAMB | ACRS STAFF |
| Pat Louden | NRC/R-III/DRP/BrS |
| MICHAEL MORGAN | NRR/DRIP/RLEP |
| Leslie Miller | NRR/DE/EMCB-C |
| DANIEL MERZKE | NRR/DRIP/RLEP |
| Ken Cheng | NRR/RLEP |
| SAMSON LEE | NRR/DRIP/RLEP |
| Jonathan Rowley | NRR/DRIP/RLEP |
| JAMES MEDOFF | NRR/DE/EMCB |
| T. KOSITY | NRR/DE/EEIB |
| NITIN PATEL | NRR/DE/EEIB |
| Ram Subbaratnam | NRR/DRIP |
| S. K. Mitra | NRR/DRIP/RLEP |
| KURT COZENS | NRR/DRIP/RLEP-B |
| RICHARD MCINTYRE | NRR/DIPM/IPSB |
| TOMMY LE | NRR/RLEP |
| HANNAH CHERNOFF | NRR/DLPM |

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**523rd FULL COMMITTEE MEETING**

**June 1-3, 2005**

**TODAY'S DATE: June 1, 2005**

**NRC STAFF - PLEASE SIGN BELOW**

**PLEASE PRINT (CLEARLY)**

| NAME | NRC ORGANIZATION |
|---|---|
| SHEILA RAY | NRR/DLPM/LPD3-1 |
| DAVID MATTHEWS | NRR/DRIP |
| Neil Ray | NRR/~~EN~~DE/EMCB |
| Mary Drouin | RES/PRAB |
| Naeem Iqbal | NRR/DSSA/SPLB |
| MARTIN STUTZKE | NRR/DSSA/SPSB |
| Greg Galletti | NRR/DIPM/IPSB |
| David Lew | RES/DRAA/PRAB |
| ~~[illegible]~~ | |
| Dave Lewis | RES/DRAA/PRAB |
| Ray Hallucci | NRR/DSSA/SPLB |
| Paul Lain | NRR/DSSA/SPLB |
| See Meng Wong | NRR/DSSA/SPSB |
| Mark Henry Salley | RES/DRAA/PRAB |
| Alexander Klein | NRR/DSSA/SPLB |
| Jocelyn Mitchell | RES |
| Gareth Parry | NRR/DSSA |
| Daniel Frumkin | NRR/DSSA/SPLB |
| Donnie Harrison | NRR/DSSA/SPSB |
| STEVEN A. LAUR | NRR/DSSA/SPSB |

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**523rd FULL COMMITTEE MEETING**

**June 1-3, 2005**

**TODAY'S DATE: June 1, 2005**

**NRC STAFF - PLEASE SIGN BELOW**

**PLEASE PRINT (CLEARLY)**

| NAME | NRC ORGANIZATION |
|---|---|
| STEPHEN DINSMORE | NRR/DSSA |
| HOSSEIN HAMZEHEE | NRC/RES/DRAA |
| JOHN C. LANE | RES/DRAA |
| Arthur Salomon | RES/DRAA/PRAB |
| Jim Lyons | NRR/DSSA |
| MIKE TSCHILTZ | NRR/DSSA/SPSB |
| ARTHUR BUSLIK | RES/DRAA/PRAB |

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**523rd FULL COMMITTEE MEETING**
**June 1-3, 2005**

**June 1, 2005**
**Today's Date**

**ATTENDEES PLEASE SIGN IN BELOW**
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| NAME | AFFILIATION |
|---|---|
| DARREL TURNER | NMC - PALISADES |
| Paul Kollath-Romano | RPI |
| Michael Heath | PGN |
| John G. Thorgersen | NMC - PBNP |
| Jenny Weil | McGraw-Hill |
| Tom King | Self |
| Blankenn | FANP |
| Susan G. Stewart | self (Duke University) |
| Francisco Joglar | SAIC |
| STEVE NOWLEN | SANDIA LAB'S |
| BIJAN NAJAFI | SAIC / EPRI |
| Bob Youngblood | ISL |
| Scott Newberry | ISL |
| [illegible] | BNL. |
| G. Martinez-Guridi | BNL |
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**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**523rd FULL COMMITTEE MEETING**
**June 1-3, 2005**

**June 1, 2005**
**Today's Date**

**ATTENDEES PLEASE SIGN IN BELOW**
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| NAME | AFFILIATION |
|---|---|
| SPYROS TRAIKOPOS | LINK |
| William D. Crouch | TVA |
| JOHN THOMAS | FIRSTENERGY |
| STEVE DORT | FIRSTENERGY |
| Ken Brune | TVA |
| Roger Jennings | TVA |
| STEVEN SCHELLIN | NMC-PBNP |
| Don Arp | TVA |
| Russell Jansen | TVA |
| Douglas Cooper | NMC. |
| Jim Schweitzer | NMC |
| Bill Herryman | NMC |
| ROBERT VINCENT | NMC |
| RICHARD GRUMBIR | AEP. |
| Douglas JOHNSON | NMC |
| BRAD FROMM | NMC |
| TODD MIELKE | NMC |
| Jim TUORE | NMC |
| Mark Ortmayer | NMC |

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**523rd FULL COMMITTEE MEETING**

**June 1-3, 2005**

**TODAY'S DATE: June 2, 2005**

**NRC STAFF - PLEASE SIGN BELOW**

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| NAME | NRC ORGANIZATION |
|---|---|
| See Meng Wong | NRR/DSSA/SPSB |
| RICHARD DIPERO | NRR/DSSA/SPLB |
| SUZANNE BLACK | NRR/DSSA |
| PAUL LAIN | NRR/DSSA/SPLB |
| Nilesh Chokshi | RES/DRAA/OERAB |
| Patrick O'Reilly | RES/DRAA/OERAB |
| RICHARD MCINTYRE | NRR/DIPM/IPSB |
| Mark Henry Salley | RES |
| Veronica Rodriguez | NRR/RLEP/A |
| SAMSON LEE | NRR/DRIP/RLEP |
| FAROUK ELTAWILA | RES/DSARE |

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**523rd FULL COMMITTEE MEETING**
**June 1-3, 2005**

**June 2, 2005**
**Today's Date**

**ATTENDEES PLEASE SIGN IN BELOW**
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| NAME | AFFILIATION |
|---|---|
| Michael Bourgeois | ENTERGY |
| George Zinke | Entergy |
| Patricia Campbell | Morgan Lewis |
| Brendan Hoffman | Public Citizen |
| Jenny Weil | McGraw-Hill |
| BIJAN NAJAFI | SAIC/EPRI |
| SPYROS TRAIFOROS | LINK |
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# ITEMS OF INTEREST

# 523rd ACRS MEETING

# JUNE 1-3, 2005

**ITEMS OF INTEREST**
**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**
**523rd MEETING**
**June 1-3, 2005**

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**STAFF REQUIREMENTS**


- Staff Requirement - Briefing on Grid Stability and Offsite Power Issues, 9:30 A.M., Tuesday April 26, 2005, Commissioners' Conference Room, One White Flint North, Rockville Maryland (Open to Public attendance) . . . . . 1

- Staff Requirement - Briefing on Status of New Site and Reactor Licensing, 9:30 A.M., Wednesday, April 6, 2005, Commissioners' Conference Room, One White Flint North, Rockville Maryland (Open to Public attendance) . . . . . 2

- Staff Requirement - Briefing on RES Programs, Performance, and Plans, 9:30 A.M., Tuesday, April 5, 2005, Commissioners' Conference Room, One Flint North, Rockville Maryland (Open to Public attendance) . . . . . 3

- Staff Requirement - Briefing on Office of Nuclear Reactor Regulation (NRR) Programs, Performance, and Plans, 1:00 P.M., Wednesday, April 20, 2005, Commissioners' Conference Room, One Flint North, Rockville Maryland (Open to Public attendance) . . 4


**SIGNIFICANT ENFORCEMENT ACTIONS**:


- EA-04-231 - Davis-Besse (FirstEnergy Nuclear Operating Company): Final Significance Determination for a White Finding and Notice of Violation (NRC Inspection Report No. 050000346/2005010 (DRS)), dated May 5, 2005 . . . . . 5-9

- EA-05-021 - Kewaunee Nuclear Power Plant Final Significance Determination for a White Finding and Notice of Violation (NRC Inspection Report No. 05000305/2004009), dated May 5, 2005, . . . . . 10-14

- EA-04-170 - LaSalle 1 (Exelon Nuclear) Notice of Violation and Proposed Imposition of Civil Penalty - $60,000 [NRC Office of Investigations Report No. 3-2004–9] and Non-Cited Violation [NRC Office of Investigations Report], May 2, 2005 . . . . . 15-19

- NOED-05-2-001 - Brunswick - Notice of Enforcement Discretion for Carolina Power & Light Company Regarding Brunswick Steam Electric Plant Unit No. 1 [TAC No. MC6977, NOED No. 05-2-001] May 17, 2005 . . . . . 20-22


**GENERIC COMMUNICATIONS**


- NRC Information Notice 2005-11: Internal Flooding/Spray-Down of Safety-Related Equipment Due to Unsealed Equipment Hatch Floor Plugs and/or Blocked Floor Drains, dated May 6, 2005 . . . . . 23-26

- NRC Regulatory Issue Summary 2005-07 - Compensatory Measures to Satisfy the Fire Protection Program Requirements, dated April 19, 2005 . . . . . 27-30


**NRC NEWS**


- "NRC to Conduct Special Inspection at Waterford Nuclear Plant," dated, May 9, 2005 . . . . . 31-32
- "NRC Announces New Director of Enforcement, dated, May 5, 2005 . . . . . 33
- "NRC Terminates Special Oversight Panel, Begins Augmented Reactor Oversight Program for Davis-Besse, dated, May 20, 2005 . . . . . 34-35
- "NRC Reports Preliminary Results of Millstone 3 Special Inspection, dated, May 18, 2005 . . . . . 36-37


**INSIDE NRC**


- Article entitled, "Performance Indicators in Oversight Program Could Use Fix, Staff Says," Volume 27/ Number 10/ May 16, 2005 . . . . . 38-41
- Article entitled, Entergy Makes Pitch to NRC Staff to Approve Vermont Yankee Uprate, Volume 27/ Number 10/ May 16, 2005 . . . . . 42-43
- Article entitled, IRSN Inaugurate Facility to Test Iodine in Severe Accidents, Volume 27/ Number 10/ May 16, 2005 . . . . . 44-45


**NUCLEAR NEWS FLASHES**


- Article entitled, U.S. News: --Investment Incentives for New Nuclear Power Plants were included....., dated May 26, 2005 . . . . . 46
- Article entitled, "Nuclear Energy is at a Crossroads, with the Next Few Months....., dated May 26, 2005 . . . . . 46
- Article entitled, "The Possible Need for a New U.K. Nuclear Pre-Licensing Regime....., dated, May 23, 2005 . . . . . 46

IN RESPONSE, PLEASE
REFER TO: M050426

May 19, 2005

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary ***/RA/***

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON GRID STABILITY AND OFFSITE POWER ISSUES, 9:30 A.M., TUESDAY, APRIL 26, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by a stakeholder panel, including representatives of the Federal Energy Regulatory Commission (FERC), the North American Electric Reliability Council (NERC), the National Association of Regulatory Utilities Commissioners, PJM Interconnection (one of the country's largest transmission system operators), and a FirstEnergy Corporation executive representing the Nuclear Energy Institute, and an NRC staff panel on grid stability and offsite power issues.

The staff should continue to (1) work with FERC, NERC, and the Department of Homeland Security on grid reliability issues; (2) communicate with stakeholders on the nuclear power plant safety aspects that can be affected by grid conditions, and encourage industry leadership to improve grid reliability; and (3) monitor and assess the reliability of the grid, and its impact on nuclear power plant safety.

At the meeting, the NRC staff discussed issuing a Temporary Instruction (TI) to NRC inspectors to gauge licensees' offsite power system readiness during the peak summer cooling season. The staff issued the TI on May 2, 2005. The TI established June 1, 2005, as the date by which the inspections are to be completed. The staff should provide the Commission offices with preliminary results of the TI inspections, including any significant corrective actions or compensatory measures taken by licensees in response to the inspections.

(EDO) (SECY Suspense: 7/29/05)

The staff should issue the final generic letter on the impact of grid reliability on plant risk no later than December 15, 2005. In addition, the staff should review NRC's programs related to operator examination and training and ensure that these programs adequately capture the importance of the influence of grid conditions and offsite power issues on the design, assessment, and safe operation of the plant, including appropriate interactions with grid operators.

The staff should evaluate the TI inspection results, licensee responses to the generic letter, and the result of the review of NRC's programs related to operator examination and training to determine whether the baseline inspection program or the operator licensing program needs to be revised to incorporate additional guidance about grid reliability, or if another round of TI inspections will be needed for next summer. If the staff determines that another TI is needed next year, the staff should perform the inspections no later than May 2006, so that licensees will have time to take corrective actions or implement compensatory measures in advance of the 2006 peak summer cooling season. The staff should inform the Commission offices of any plans to revise the baseline inspection program, the operator licensing program, or the need to issue another TI on grid reliability for 2006, no later than April 28, 2006.

cc: Chairman Diaz
Commissioner McGaffigan

IN RESPONSE, PLEASE
REFER TO: M050406

May 10, 2005

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON STATUS OF NEW SITE AND REACTOR LICENSING, 9:30 A.M., WEDNESDAY, APRIL 6, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the status of new site and reactor licensing. The following items were identified for staff action:

1. The staff should engage the Nuclear Energy Institute and potential applicants to establish better estimates for potential new reactor deployment (e.g., in order to develop realistic schedule models and resource estimates). The Congress should be kept informed about new reactor activities and the impact on the NRC budget. In accordance with Commission policy to support design certification reviews before a construction permit or combined license (COL) has been filed, the staff should plan on developing budget estimates to conduct all pre-application reviews, design certification reviews, early site permit reviews, and COL reviews which are reasonably expected to be submitted in FY2007 or beyond (i.e., where the uncertainty associated with receiving the request for review is low to medium). However, when it comes time to execute the budget in FY2007 and beyond, the staff should give priority to applications that are aligned with a COL partner, and assign a lower priority to applications that are not clearly aligned with a COL partner, if funding is not approved to review all applications. This practice will improve the reliability of our budget process and the discharge of both our licensing and regulatory oversight obligations.

2. The Office of the Secretary, using the Commission agenda planning process, will schedule another meeting on this topic for the November 2005 timeframe. The meeting should focus primarily on the role of inspections, tests, analyses, and acceptance criteria (ITAAC) and on the status of the new construction inspection program (CIP).

3. In light of the impending retirement of key senior staff over the next few years, applicable sections of the Standard Review Plan (NUREG 0800), other guidance documents, and office procedures need to be revised in a timely manner, to ensure up-to-date guidance is available for the next generation of staff that will be responsible for reviewing and licensing new sites and new reactors. NRR should develop an integrated and continuing plan for updating licensing review guidance and provide the plan, along with a schedule for completion, to the Commission. The staff should prioritize the list of guidance documents that need revision, and focus on updating the most critical items first.

(EDO) (SECY Suspense: 12/30/05)

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO

IN RESPONSE, PLEASE
REFER TO: M050405

May 9, 2005

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary ***/RA/***

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON RES PROGRAMS, PERFORMANCE, AND PLANS, 9:30 A.M., TUESDAY, APRIL 5, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the Office of Nuclear Regulatory Research (RES) programs, performance, and plans.

The staff should look at key areas where NRC can improve collaboration with international and national research programs on issues like low dose radiation analysis and consequence analysis.

The staff should ensure that NRC research programs are as realistic as practicable. Bounding analyses should be clearly labeled as such and the staff should explain why bounding analyses are adequate for regulatory decisions where they are used.

The RES staff should work with NRR to develop a formal program plan to make a risk-informed and performance-based revision to 10 CFR Part 50, including revisions to the applicable Regulatory Guides, Standard Review Plans, or other guidance documents.

The staff should reevaluate its contract management activities to assure that guidance to contractors is well documented and deliverables are specified in detail. Past performance information should be considered in making award decisions for agreements with DOE laboratories and other federal agencies, consistent with the current practice for commercial contracts.

The staff should continue human reliability research to help reduce the abnormal occurrences in the materials area with respect to misuse of radioisotopes and radioactive material in medical and industrial applications consistent with existing user needs.

The Commission is particularly interested in staff efforts to identify areas in which fire modeling knowledge needs to be advanced to support regulatory decisionmaking, particularly in the area of the significance determination process for fire issues.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OIG
OGC
CFO
OCA

IN RESPONSE, PLEASE
REFER TO: M050420B

May 3, 2005

MEMORANDUM FOR: Luis A. Reyes
Executive Director for Operations

FROM: Andrew L. Bates, Acting Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON OFFICE OF NUCLEAR REACTOR REGULATION (NRR) PROGRAMS, PERFORMANCE, AND PLANS, 1:00 P.M., WEDNESDAY, APRIL 20, 2005, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the status of the Office of Nuclear Reactor Regulation (NRR) programs, performance, and plans.

The staff should continue its efforts to bring to closure important issues in the area of fire protection.

cc: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
OGC
CFO
OCA
OIG
DOC
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

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*Last revised Wednesday, May 04, 2005*

# EA-04-231 - Davis-Besse (FirstEnergy Nuclear Operating Company)

May 5, 2005

EA-04-231
EA-04-232

Mr. Mark B. Bezilla
Vice President-Nuclear, Davis-Besse
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT
NO. 05000346/2005010(DRS)

Dear Mr. Bezilla:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary White finding identified in Inspection Report No. 05000346/2004018(DRS). The inspection finding was assessed using the Significance Determination Process (SDP) and was preliminarily characterized as White (i.e., a finding with low to moderate increased importance to safety, which may require additional NRC inspections) because it was associated with a failure to implement a risk significant planning standard (RSPS). This preliminary White finding concerned the failure of the emergency planning zone (EPZ) sirens identified on May 7, 2004.

The preliminary White finding was associated with the Ottawa County Sheriff's Office loss of its capability to activate 49 EPZ sirens, located in Ottawa County, between April 27 and May 7, 2004, and potential vulnerability to unknowingly lose its capability to activate the 49 EPZ sirens located in Ottawa County for approximately 30 days prior to a routine Alert and Notification System (ANS) test conducted on May 7, 2004. The preliminary White finding was also associated with an apparent violation of 10 CFR 50.47(b)(5), "Emergency Plans."

In our letter to you dated January 13, 2005, transmitting the inspection report and preliminary significance determination, we provided FirstEnergy Nuclear Operating Company (FENOC) an opportunity to request a Regulatory Conference or provide a written response. You declined the opportunity to discuss this issue in a Regulatory Conference and instead, on February 14, 2005, provided a written response. A copy of the your written response to the NRC has been entered in the NRC's document system (ADAMS) and is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, ADAMS Accession Number ML050470307.

In your letter, you disagreed with the NRC's application of the SDP and preliminary significance assessment of the finding, and provided information that you requested the NRC use in a re-evaluation of the finding. Specifically, you indicated that the significance of the finding should be assessed assuming the sirens were unavailable for only 10 days and with consideration that an alternate method was available to notify the public of an emergency. You also disagreed with our statement that you had several opportunities to correct the ANS performance indicator (PI) data prior to submitting the data to the NRC, including information provided to you by the NRC. Finally, you noted that the apparent violation associated with the submittal of the incorrect ANS PI data appeared to be consistent with the Severity Level IV violation examples

included in the NRC's Enforcement Policy.

With regard to the length of time the sirens were unavailable, we re-examined the inspection results and concluded that the county officials lost the capability to activate all 49 Ottawa County EPZ sirens for approximately 10 days. In addition, we determined that the potential existed for the county officials to lose the capability to activate the 49 Ottawa County EPZ sirens for approximately 30 days. Since the ANS is designed to permit the county officials to activate the system, we determined that a loss of this capability would constitute a failure versus a degradation of the system. In addition, we concluded that the system should be considered degraded for that period of time when the potential existed for the system to be made inoperable due to a design flaw and a deficiency in the maintenance program. Therefore, we determined that the county officials' inability to activate the 49 Ottawa County EPZ sirens for 10 days and potential loss of its capability to activate the 49 Ottawa County EPZ sirens for 30 days, was appropriately considered to be a degradation of the Risk-Significant Planning Standard and was properly characterized as a White finding.

During our initial significance assessment of the finding, we did not specifically consider the availability of route alerting as an alternate method for public notification during an emergency. However, we note that the SDP already includes consideration of the presence of an alternate notification system. As a result, we concluded that a re-assessment of the significance of the finding, considering the availability of an alternate public notification method, was not necessary. We also determined that, had route alerting not been available, the finding may have been characterized as being greater than a White finding.

With regard to the statements in our January 2005 letter associated with your submittal of incorrect ANS PI data, we reviewed the information you provided in your response and determined that sufficient information was available to your staff, prior to your submittal of the discrepant ANS PI data, to indicate that the additional silent tests of the EPZ sirens should not have been included in the ANS PI data. Specifically, we determined that Nuclear Energy Institute (NEI) publication NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," an NRC-endorsed document issued in November 2001, provided clear guidance which would preclude inclusion of the silent tests of the EPZ sirens in the ANS PI data. The NRC also raised questions regarding the validity of your changing the testing methodology, in the middle of a testing period, and the need for Federal Emergency Management Agency (FEMA) approval for the methodology change. Although the NRC did not raise a specific concern regarding which equipment was used to conduct the silent tests of the EPZ sirens until after the ANS PI data were submitted in July 2004, we determined that sufficient guidance was available and questions were raised regarding the revised methodology, which should have resulted in your staff determining that the silent tests of the EPZ siren tests should not be included in the ANS PI data.

After considering the information developed during the inspection and the additional information you provided in your February 14, 2005, letter, the NRC has concluded that the inspection finding associated with the degradation of the EPZ sirens is appropriately characterized as White (i.e., an issue with low to moderate increased importance to safety, which may require additional NRC inspections).

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC has also determined that the failure to ensure the means to provide early notification and clear instruction to the populace within the plume exposure pathway EPZ is a violation of 10 CFR 50.47(b)(5), as cited in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, the Notice of Violation is considered an escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

Finally, in your February 14, 2005, letter, you agreed that the 10 CFR 50.9(a) violation of the Commission's regulations, documented in our January 2005 letter to you and associated with the submittal of discrepant ANS PI data for the second and third calendar quarters of 2004, was appropriately characterized as a Severity Level IV violation in accordance with the NRC's Enforcement Policy. On November 4, 2004, you submitted a letter to the NRC correcting the performance indicator data.

Based on the results of this inspection, we have determined that your submittal of discrepant ANS PI data for the second and third quarters of 2004 is a Severity Level IV violation of 10 CFR 50.9, "Completeness and Accuracy of Information." You

documented the NRC's concerns regarding your submittal of the ANS PI data in your corrective action program as Condition Report 04-06632, the violation was not willful, and compliance was restored within a reasonable period of time. Therefore, this violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the Enforcement Policy. The NCV was described as Apparent Violation 05000346/2004018-01 in NRC Inspection Report No. 05000346/2004018(DRS). If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this letter, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and the NRC Resident Inspector at the Davis-Besse facility (EA-04-232).

For the entire inspection period, the Davis-Besse Nuclear Power Station was under the Inspection Manual Chapter (IMC) 0350 Process. The Davis-Besse Oversight Panel assessed the inspection findings and other performance data to determine the required level and focus of followup inspection activities and any other appropriate regulatory actions. Even though the Reactor Oversight Process had been suspended at the Davis-Besse Nuclear Power Station, it was used as guidance for inspection activities and to assess findings. Accordingly, we will use the NRC Action Matrix, in accordance with IMC 0305, to determine the most appropriate NRC response for this event. We will notify you, by separate correspondence, of that determination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select What We Do, Enforcement, then Significant Enforcement Actions.

Sincerely,

**/RA by Mark Satorius Acting for/**

James L. Caldwell
Regional Administrators

Docket No. 50-346
License No. NPF-3

Enclosure: Notice of Violation

cc w/encl: The Honorable Dennis Kucinich
G. Leidich, President - FENOC
J. Hagan, Senior Vice President
Engineering and Services, FENOC
L. Myers, Chief Operating Officer, FENOC
Plant Manager
Manager - Regulatory Compliance
M. O'Reilly, Attorney, FirstEnergy
Ohio State Liaison Officer
R. Owen, Administrator, Ohio Department of Health
Public Utilities Commission of Ohio
President, Board of County Commissioners
of Lucas County
J. Papcun, President, Ottawa County Board of Commissioners
W. King, Federal Emergency Management Agency, Region V

During an NRC inspection conducted between October 25 and October 29, 2004, at the Davis-Besse Nuclear Power Station, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10 CFR 50.54(q) requires, in part, that a licensee authorized to operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in Section 50.47(b). Title 10 CFR 50.47(b)(5) requires, in part, the means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone (EPZ) have been established.
>
> Section 7.7 of Revision 23 of the Davis-Besse Nuclear Power Station's emergency plan indicated that the EPZ "Prompt Notification System" implements the requirements of 10 CFR 50.47(b)(5). Section 7.7 states that the Alert and Notification System (ANS), consists of 54 sirens that were installed to provide an acoustic alerting signal for the residents and transients within the 10 mile radius of the Davis-Besse Station, and that the sounding of these sirens would alert the public to tune to local radio stations for Emergency Alert Station messages. The emergency plan also indicated that local officials would activate the sirens from the Ottawa County Sheriff's Office.
>
> Contrary to the above, between April 27 and May 7, 2004, the FirstEnergy Nuclear Operating Company, a licensee authorized to operate the Davis-Besse Nuclear Power Station, failed to provide a means for early notification and clear instruction to the populace within the plume exposure pathway EPZ. Specifically, on April 6th, the time signature of the siren activation equipment, located in the Ottawa County Sheriff's Office, was not revised the following the change to daylight savings time. On April 26th, the licensee conducted a maintenance activity which incorrectly set the time signature of one siren to one hour ahead of the actual day light savings time. On April 27th, the licensee conducted a routine polling of all of the EPZ sirens which caused the time signatures of the remaining EPZ sirens to be set one hour ahead of the actual daylight savings time. The combination of these changes caused the time signatures of the activation equipment and the EPZ sirens to differ by more than the allowed 1.5 hours which blocked activation of the sirens. As a result, the Ottawa County Sheriff's Office could not activate the 49 EPZ sirens located in Ottawa County until May 7, 2004, when the combined results of these events were identified.

This violation is associated with a White Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, FirstEnergy Nuclear Operating Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-04-231," and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid further violations; and (4) the date when full compliance will be achieved. Your response may reference or include
previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an Order 9 or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential

commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 5th day of May 2005

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*Last revised Friday, May 06, 2005*

# EA-05-021 - Kewaunee (Nuclear Management Company, LLC)

May 5, 2005

EA-05-021

Mr. Craig W. Lambert
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 Hwy 42
Kewaunee, WI 54216-9511

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT
FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT 05000305/2004009)

Dear Mr. Lambert:

The purpose of this letter is to provide you with the final results of our significance determination of a finding which was described in Inspection Report 05000305/2004009, issued February 14, 2005, that involved the inability of your staff to rapidly close the containment equipment hatch during cold shutdown conditions due to an interference. The interference was caused by the inadequate design of a rail system that was installed in the containment to facilitate reactor vessel head replacement activities. This finding was assessed using the Significance Determination Process (SDP) and was preliminarily characterized as White (i.e., a finding with low to moderate increased importance to safety, which may require additional NRC inspection).

In our letter dated February 18, 2005, the Nuclear Regulatory Commission (NRC) provided the Nuclear Management Company (NMC) an opportunity to either request a Regulatory Conference to discuss this finding, or to explain your position in a written response. At your request, a Regulatory Conference was scheduled and held on March 17, 2005, to discuss your views on this issue. Supporting documentation for the conference was submitted by you in letters dated March 12 and March 13, 2005. The Regulatory Conference summary, including presentation slides, is available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of NRC's document system (ADAMS) at accession number ML050620590. The NRC's document system is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

At the Regulatory Conference, you presented an overview of the event, related corrective actions, and the methodology and results of your independent safety assessment of the preliminary White finding, including the risk analysis results that were used in your independent safety assessment. In your presentation, you stated that it was your view that the finding would be more properly characterized as Green (i.e., a finding of very low safety significance). The NRC and NMC staffs held extensive discussions regarding technical issues related to your analysis. Topics discussed included the probability of closing the containment hatch, if there was a station blackout event, the assumed mission times for the emergency diesel generator (EDG) and the technical support center diesel generator (TSC/DG), the reliability of the TSC/DG, and the exposure time for the finding. Specific topics that affected the final significance determination are discussed in the following paragraphs. Subsequent to the conference, on March 23, 2005, you submitted a written response to certain questions that were asked at the conference. This information was also considered in our final significance determination as described below.

Regarding the probability of closing the containment hatch during a postulated station blackout event, you presented a timeline for closing the hatch, a containment habitability analysis, and a human error probability for the failure to close the hatch. The NRC reviewed your hatch closure timeline and determined that a high degree of uncertainty existed in the time estimates for the decisions and tasks required to close the hatch. In particular, you estimated that the time from the discovery of the interference during the postulated station blackout event to the decision on the method for clearing the hatch interference to be 25.5 minutes, with an additional 40 minutes for the actual clearing of the interference. This and other aspects of the timeline appear to be overly optimistic given that an emergency condition would exist, the hatch interference was not previously known, and there was no existing method or procedure with specific instructions to remove the interference in place prior to discovery of the interference. Additionally, in contrast to the timeline presented, when this interference was identified in the plant, almost 7 hours were required to remove the interference. The containment habitability analysis supported the timeline that you established. However, because of the uncertainty in the timeline, containment habitability is also uncertain for any timeline that would exceed the one that was presented.

The timeline and the results of the containment habitability analysis were used in your calculation of the hatch closure failure probability, which was estimated at 3.7E-1. You stated that your result was based on assuming the decision making for developing the removal method of the hatch interference was a skill-based activity. We determined that the assumption that the activity was skill-based was not justified because the obstruction was not previously known to plant personnel and, in fact, the inner rail system was intended to be designed not to interfere with the equipment hatch door. The discovery of the interference would have occurred only when the closure was necessary, requiring personnel to develop an action plan and execute that plan during emergency conditions. Therefore, the NRC determined that the decision making for developing the removal method of the hatch was a knowledge-based activity.

Given the large uncertainty in your estimated hatch closure timeline and the NRC determination that the removal of the interference would be a knowledge-based rather than a skill-based activity, the NRC concluded that use of your hatch closure failure probability was not justified. The NRC performed an evaluation of the human error probability (HEP) for removing the interference and closing the hatch using the NRC's Standardized Plant Analysis Risk (SPAR) method. The NRC assumed that no procedures with specific instructions existed for the identification and removal of the interference and that extreme stress would be present. The estimated HEP was 1.0. This estimation assumes that adequate time is available and that the environment would not prohibit the actions. Due to the uncertainty with your timeline for hatch closure, the NRC does not consider these assumptions to be accurate. However, the effect on the risk estimate would be unchanged since the HEP would also be 1.0 if there was either inadequate time available or if the containment habitability would not support the required actions.

This HEP result is consistent with the NRC's established practice for SDP analyses. Phase 2 analyses allow limited credit for recovery actions if the actions are proceduralized, the staff are trained, enough time is available, and the environment for performing the actions would remain habitable. In Phase 3 of the SDP, the SPAR method explicitly models the effects of various performance shaping factors including the quality of procedures, the environment, and the stress level involved in performing the actions.

For the EDG and TSC/DG mission times, you stated that the 1.6 hours used represented the probability weighted average loss of offsite power recovery time. The NRC calculated the average loss of offsite power recovery time using data from actual loss of offsite power events published in available draft reports, "Evaluation of Loss of Offsite Power Events: 1986 - 2003," and "Station Blackout Risk Evaluation for Nuclear Power Plants," and determined that the mission time used for this assessment should be 5.6 hours. For a sensitivity analysis, the NRC also used 24 hours, which is consistent with the mission time used in the NRC's SPAR models.

In your analysis, you stated that the TSC/DG failure to run probability was based on a Bayesian update of generic data from NUREG/CR-5500 Volume 5, "Emergency Diesel Generator Power System Reliability 1197 - 2003," with plant specific data. We determined that the use of the EDG data for updating the TSC/DG was not appropriate because the EDG reliability study did not include diesel generators used for station blackout conditions similar to the TSC/DG. Therefore, the NRC concluded that the failure to run estimate for the TSC/DG should be based only on the plant-specific data, which resulted in a failure to run estimate of 5.2E-3/hr. Also, you did not include a term for maintenance unavailability of the TSC/DG in your overall failure probability. During the Regulatory Conference you stated that TSC/DG maintenance was typically an online maintenance activity that would not be performed during an outage. The NRC asked if the TSC/DG had ever been unavailable due to maintenance during an outage. In your reply, dated March 23, 2005, you provided information that the TSC/DG had, in a prior outage, been taken out of service for maintenance. Therefore, the NRC determined that maintenance unavailability of the TSC/DG should be considered in the overall failure probability. However, the maintenance unavailability was small compared to the other possible failure modes of the TSC/DG (including human error) and so the result of the analysis remains unchanged with the addition of maintenance unavailability.

The exposure time, or the time that the containment hatch obstruction existed concurrent with an unavailable "A" EDG, used in your analysis was 2.73 days. This did not include a 3.7-hour period of unavailability of the "B" EDG. You indicated that the reason stated for not including this exposure time was that the out of service time was less than the time to core uncovery. Additionally, your analysis included credit for recovery of a failed EDG. The NRC determined that it was appropriate to consider the 3.7-hour period of unavailability of the "B" EDG given that there is no assurance that the return to service of the EDG from the maintenance would always occur in that time period and because recovery of a failed EDG was explicitly modeled in the analysis. However, similar to the TSC/DG maintenance unavailability, the inclusion of this additional period of exposure time did not significantly influence the result of our analysis.

We reviewed all other aspects of your independent safety assessment of this issue and determined that, with the exception of the specific topics discussed in this letter, your assumptions were reasonable.

After considering the information developed during the inspection, the supporting information you provided prior to the Regulatory Conference, the information you provided at the Regulatory Conference, and your written response to our questions at the Regulatory Conference, the NRC concluded through additional independent risk calculations that the change in the Large Early Release Frequency associated with the inspection finding was still greater than 1E-7 and was appropriately characterized as White (i.e., an issue with low to moderate safety significance which may require additional NRC inspection).

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual 0609, Attachment 2. Appeals to reduce the significance of an inspection finding will be considered as having sufficient merit for review by this appeal process only if the contention falls into one of the following categories: (1) actual (verifiable) plant hardware, procedures, or equipment configurations were not considered by the staff; or (2) the staff's significance determination process was inconsistent with the applicable SDP guidance or lacked justification.

The NRC has also determined that the inadequate design of the rail system that was installed in the containment to facilitate the reactor vessel head replacement activities and the failure to have adequate procedures with specific instructions for rapid removal of the interior rail to allow expeditious hatch closure, is a violation of 10 CFR Part 50, Appendix B, Criterion V, as cited in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in Inspection Report 05000305/2004009. In accordance with the NRC Enforcement Policy, the Notice is considered escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

Because plant performance for this issue has been determined to be in the regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for these events. We will notify you, by separate correspondence, of that determination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARs) component of the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (The Public Electronic Reading Room).

Sincerely,

**/RA by Mark Satorius Acting for/**

James L. Caldwell
Regional Administrator

Docket No. 50-305
License No. DPR-43

Enclosure: Notice of Violation

cc w/encl: J. Cowan, Executive Vice President,
Chief Nuclear Officer
Plant Manager
Manager, Regulatory Affairs
J. Rogoff, Vice President, Counsel & Secretary
D. Molzahn, Nuclear Asset Manager,
Wisconsin Public Service Corporation
L. Weyers, Chairman, President and CEO,
Wisconsin Public Service Corporation
D. Zellner, Chairman, Town of Carlton
J. Kitsembel, Public Service Commission of Wisconsin

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NOTICE OF VIOLATION

Nuclear Management Company
Kewaunee Nuclear Power Plant

Docket No. 50-305
License No. DPR-43
EA-05-021

During an NRC inspection conducted from October 1 through December 31, 2004, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

Title 10 CFR Part 50, Appendix B, Criterion V, (Instructions, Procedures, and Drawings) requires, in part, that activities affecting quality be prescribed by documented instructions, or procedures of the type appropriate to the circumstances and shall be accomplished in accordance with these instructions, or procedures.

Kewaunee Nuclear Power Plant Procedure CMP-89 A-02, "Containment Building Inner Equipment Door Opening and Closing Instructions," a procedure affecting quality, required that any equipment which passes through and could obstruct containment hatch closure be designed to allow rapid removal in order to ensure expeditious containment building equipment hatch closure should it become necessary to do so.

Contrary to the above, on October 11, 2004, the licensee installed in the containment an interior steel runway track, equipment which could obstruct the containment hatch closure, which was not designed to allow rapid removal in order to ensure expeditious containment building equipment hatch closure. Specifically, the design of the interior steel runway track obstructed the containment hatch closure and could not be rapidly removed in order to permit an expeditious closure of the containment hatch should it become necessary to do so. In addition, the licensee did not develop or have in place procedures or plans to effect a rapid removal of the interior portion of the steel rail system to eliminate the interference.

This violation is associated with a White Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, the Nuclear Management Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the Kewaunee Nuclear Power Plant, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-05-021" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be withheld and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 5th day of May 2005

Privacy Policy | Site Disclaimer
*Last revised Friday, May 06, 2005*

# EA-04-170 - LaSalle 1 (Exelon Nuclear)

May 2, 2005

EA-04-170

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $60,000 [NRC OFFICE OF INVESTIGATIONS REPORT NO. 3-2004-009] AND NON-CITED VIOLATION [NRC OFFICE OF INVESTIGATIONS REPORT

Dear Mr. Crane:

Th[illegible]ers to information provided to the U.S. Nuclear Regulatory Commission (NRC) by Exelon Nuclear (Exelon) on January 29, 2004, indicating that four contract employees improperly entered a high radiation area at the LaSalle County Station (LaSalle) on January 25, 2004. The NRC Office of Investigations (OI) investigated the matter and concluded that three contract employees willfully violated radiation protection procedures associated with entry into high radiation areas (HRA).

In a letter dated November 19, 2004, transmitting the Summary of Investigation associated with OI Report No. 3-2004-009, we provided you an opportunity to address the apparent violations identified in the letter and previously documented in NRC Inspection Report Nos. 05000373/2004002 and 05000374/2004002 by either attending a pre-decisional enforcement conference or by providing a written response before we made our final enforcement decision. In a letter dated December 17, 2004, you provided a response to the apparent violations.

Based on the information developed during the NRC inspection and investigation and based on information that you provided in your December 17, 2004, letter, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in the subject inspection and OI investigation report. In summary, on January 25, 2004, a contract foreman and three contract workers were assigned to conduct outage work associated with a valve located in the reactor building. In preparation for the work, the foreman signed in on a radiation work permit (RWP) associated with entry into HRAs located in the turbine and auxiliary buildings but not for entry into the Unit 1 reactor building raceway described below. The three contract workers signed in on a pre-outage RWP associated with minor maintenance activities which did not permit entry into HRAs. During a walk down of the work, the contract workers could not locate the valve, and the foreman took the contract workers into a posted HRA in the Unit 1 reactor building raceway to locate the valve. Prior to entering the HRA, at least one of the contract workers told the foreman that the contract workers were not signed in on an RWP that permitted entry into HRAs. Before entering the HRA, two of the contract workers were aware that they had not received a briefing by radiation p[illegible]ion personnel for the HRA, a prerequisite for entry into HRAs. Therefore, the actions of the foreman and two c[illegible]tor workers[1] are considered a willful violation, representing careless disregard of requirements, and the violation has been categorized in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600 at Severity Level III.

In a December 17, 2004, letter, you acknowledged that a willful violation occurred, presented information indicating that the violation should be categorized at Severity Level IV, and indicated that you believed that the violation met the NRC criteria to be categorized as a Non-Cited Violation (NCV). Specifically, you contended that the foreman was a low-level in[illegible]al and should not be considered a licensee official as defined in Section IV.A of the Enforcement Policy. You also re[illegible]ced three previous NRC enforcement actions [2] involving a first-line supervisor willfully causing an NRC licensee to be in violation of regulatory requirements where each prior action was individually categorized as a Severity Level IV violation. In making our decision, we determined that the foreman at LaSalle was responsible for the actions of himself and the contractor employees and that the foreman willfully failed to ensure that both he and the three employees complied with the licensee's radiation protection procedures associated with entry into an HRA. Two of the workers also were aware that entry into an HRA was prohibited prior to completing the licensee's radiation protection procedural requirements.

Regarding the three prior enforcement cases referenced in your letter, we determined that each of the cases involved the willful actions of a supervisor; however, none caused other individuals to violate licensee procedures. In the current case, the foreman's actions caused three other individuals to violate licensee procedures. Additionally, two contract workers were aware of licensee prerequisites for entry into an HRA but did not complete the requirements prior to entering the HRA. As a result, this violation cannot be considered an isolated act of an employee since four employees, one of whom was a foreman, were involved in the violation. After reviewing the information presented in your December 17, 2004, letter and the Enforcement Policy, the NRC concluded that you did not provide a sufficient basis for the NRC to re-categorize the violation at Severity Level IV or issue an NCV. Therefore, this willful violation is appropriately categorized, in accordance with the Enforcement Policy, as a Severity Level III violation.

In accordance with the Enforcement Policy, a base civil penalty in the amount of $60,000 is considered for a Severity Level III violation. Because the violation was willful, the NRC considered whether credit was warranted for the civil penalty adjustment factors of *Identification* and *Corrective Action* in accordance with the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy.

The violation was self-revealed through an event, an alarming dosimeter. While a licensee can be given credit for the *Identification* civil penalty adjustment factor in some circumstances for a self-revealing event, the Enforcement Policy also provides that consideration must be given to, among other things, prior opportunities to identify the problem requiring co[illegible]tive action and a licensee's efforts to determining the root cause. On balance, credit is not warranted for the civil p[illegible] adjustment factor for *Identification* as three, similar, non-willful violations of radiation protection requirements oc[illegible]d during the period of December 30, 2003, to January 21, 2004. Specifically, on December 30, 2003, contract workers entered an HRA in the 1A turbine-driven reactor feedwater pump room without receiving a briefing from radiation protection personnel (see also the discussion below). On January 20, 2004, two technicians entered the 1B residual heat removal room HRA without receiving a pre-job briefing from radiation protection personnel and without reviewing and signing the RWP for that area. On January 21, 2004, a craft worker entered the Unit 1 heater bay HRA and had not received a pre-job briefing from radiation protection personnel prior to entering that HRA. These violations constituted sufficient prior notice that employees were not following radiation protection procedures and demonstrated that your staff did not fully address the root cause for the continuing violations.

Credit was warranted for the *Corrective Action* civil penalty adjustment factor. Corrective actions consisted of but were not limited to the following: (1) implementing a requirement for all workers to stop at a radiation protection desk to receive a radiation protection briefing prior to logging onto an RWP that allowed access to an HRA; (2) adding radiation protection control points to challenge workers on their understanding of RWP requirements, HRA compliance, work area understanding, and response to dosimetry alarms; (3) revising the initial radiation worker training to highlight HRA entry requirements and to add worker acknowledgment warnings on computer screens during the access control process; and (4) requiring all transient refueling outage workers to attend and pass a dynamic learning activity on proper HRA entry. Additionally, the contractor emphasized radiological and personnel safety to its craft workers and assigned employees to monitor the preparedness of workers to enter into HRAs.

Therefore, to emphasize the significance of willful violations and the need for prompt identification of and corrective action for violations, I have been authorized, after consultation with the Director, Office of Enforcement, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalty (Notice) in the base amount of $60,000 for the Severity Level III violation.

T[illegible]C has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation, and the date when full compliance was achieved, is already addressed on the docket in your December 17, 2004, letter. Therefore, you are not required to respond to the provisions of 10 CFR 2.201 unless the description in your letter does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional

information, you should follow the instructions specified in the enclosed Notice. However, you are required to either pay the proposed civil penalty or respond in accordance with the instructions in the enclosed Notice.

If you disagree with this enforcement action, you may request alternative dispute resolution (ADR) with the NRC in an attempt to resolve this issue. Alternative Dispute Resolution is a general term encompassing various techniques for resolving conflict outside of court using a neutral third party. The technique that the NRC has decided to employ during a pilot program which is now in effect is mediation. Additional information concerning the NRC's pilot program is described in the enclosed brochure (NUREG/BR-0317). Additional copies can be obtained at http://www.nrc.gov/what-we-do/regulatory/enforcement/adr.html. The Institute on Conflict Resolution (ICR) at Cornell University has agreed to facilitate the NRC's program as an intake neutral. Please contact ICR at 877-733-9415 within 10 days of the date of this letter if you are interested in pursing resolution of this issue through ADR. You may also contact Mr. Nick Hilton, Office of Enforcement, at 301-415-3055 for additional information.

With regard to the prior non-willful violations described above for application of the civil penalty adjustment factor associated with *Identification*, two issues were dispositioned as NCVs in Inspection Report Nos. 05000373/2004002 and 05000374/2004002 as NCV Nos. 05000373/2004-004 (event on January 20, 2004) and 05000373/2004-005 (event on January 21, 2004). NRC OI also investigated the similar issue that occurred on December 30, 2003, when three contract workers entered a posted HRA in the 1A turbine-driven reactor feedwater pump room (OI Report No. 3-2004-005, synopsis enclosed), and concluded that a willful violation of NRC requirements did not occur in that instance. The issue was also described in Inspection Report Nos. 05000373/2004002 and 05000374/2004002 as Unresolved Item No. 05000373/2004002-03. The NRC concluded that the actions of contract workers on December 30, 2003, represented a violation of the LaSalle Technical Specification 5.4.1.b and Section 4.7 of Exelon Procedure No. RP-AA-460, "Controls for High and Very High Radiation Areas." This was evaluated under the risk significance determination process as having very low safety significance (Green). Since the violation was identified by Exelon and entered into the corrective action system as Condition Report No. 192902, the violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy. If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this letter, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, Region III, and the NRC Resident Inspector at the LaSalle County Station.

If you have any questions, please contact Steven Orth, Health Physics Team Leader at 630-829-9827.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS) accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response, if you choose to respond, should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov: select **What We Do, Enforcement,** then **Significant Enforcement Actions.**

Sincerely,

**/RA by Geoffrey Grant Acting for/**

James L. Caldwell
Regional Administrator

Docket No. 50-373
License No. NPF-11

Enclosures:
1. Notice of Violation and Proposed Imposition of Civil Penalty
2. Synopsis of OI Report No. 3-2004-005
3. NUREG/BR-0254, Payment Methods (Licensee only)
4. NUREG/BR-0317, Post-Investigation ADR Program (Licensee only)

cc w/ encls 1 and 2:

Site Vice President - LaSalle County Station

LaSalle County Station Plant Manager
Regulatory Assurance Manager - LaSalle County Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional
Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional
Operating Group
Manager Licensing - Clinton and LaSalle
Senior Counsel, Nuclear, Mid-West Regional
Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

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NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Exelon Nuclear
LaSalle County Station, Unit 1

Docket No. 50-373
License No. NPF-11
EA-04-170

During an NRC investigation completed on July 27, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the NRC proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violation and associated civil penalty are set forth below:

LaSalle County Station Technical Specification 5.4.1.a provides, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

Section 7.e.(1) of Appendix A to Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)," Revision 2, February 1978, provides, in part, that the licensee establish written procedures to control access to radiation areas including a radiation work permit (RWP) system.

Section 4.7. of Exelon Nuclear Procedure No. RP-AA-460, "Controls for High and Very High Radiation Areas," Revision 4, December 2003, a procedure that implements Regulatory Guide 1.33, provides, in part, that an individual requesting entry into a high radiation area (HRA) review survey data for the applicable area, review and sign the appropriate RWP, and receive a briefing from radiation protection personnel concerning dose rate and low dose area information, tasks allowed to be performed in the area, required dosimetry, alarm set points and maximum stay-times, and proper control of barricades and postings upon entering and exiting the area.

Contrary to the above, on January 25, 2004, four employees (a foreman and three contract workers) of The Venture, a contractor at the LaSalle County Station, entered a posted HRA on the 694-foot elevation of the Unit 1 reactor building raceway, and each employee failed to review the survey data for the specific HRA, review and sign the RWP appropriate to the area, and receive a briefing from radiation protection personnel prior to entering the HRA.

This is a Severity Level III violation (Supplement IV).
Civil Penalty - $60,000. (EA-04-170)

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation, and the date when full compliance was achieved, is already addressed on the docket in your December 17, 2004, letter. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation, EA-04-170," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the LaSalle County Station, within 30 days of the date of the letter transmitting this Notice of Violation (Notice).

The licensee may pay the civil penalty proposed above in accordance with NUREG/BR-0254 and by submitting to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, a statement indicating when and by what method payment was made, or may protest imposition of the civil penalty in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the licensee fail to answer within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalty, an order imposing the civil penalty will be issued. Should the licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation, EA-04-170," and may: (1) deny the violation listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the factors addressed in Section VI.C.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201 but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the licensee is directed to the other provisions of 10 CFR 2.205 regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, statement as to payment of civil penalty, and Answer to a Notice of Violation) should be addressed to: Frank J. Congel, Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738, with a copy to the Regional Administrator and Enforcement Officer, U.S. Nuclear Regulatory Commission, Region III, and a copy to the NRC Resident Inspector at the LaSalle County Station.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS) which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction.

In accordance with 10 CFR 19.11, you may be required to post this Notice within 2 working days.

Dated this 2nd day of May 2005

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1. Information developed by OI indicated that a third contract worker improperly entered the HRA at the same time as the others; however, OI concluded that the actions of the third contractor were not willful.

2. The previous enforcement actions referenced by the licensee were: (1) EA-00-057 at Duane Arnold Energy Center, (2) EA-00-075 at the Braidwood Station, and (3) EA-03-153 at the Beaver Valley Power Station.

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*Last revised Thursday, May 05, 2005*

# NOED-05-2-001 - Brunswick (Carolina Power and Light Company)

May 17, 2005

Carolina Power and Light Company
ATTN: Mr. C. J. Gannon
Vice President
Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461

SUBJECT: NOTICE OF ENFORCEMENT DISCRETION FOR CAROLINA POWER & LIGHT REGARDING BRUNSWICK STEAM ELECTRIC PLANT UNIT NO. 1 [TAC NO. MC6977, NOED NO. 05-2-001]

By letter dated May 13, 2005, you requested that the NRC exercise discretion to not enforce compliance with the actions required in Brunswick Steam Electric Plant, Unit 1 Technical Specification (TS) Limiting Condition for Operation (LCO) 3.0.3. Entry into this LCO was based on the failure of Unit 1 to meet the requirements of Action Condition D of LCO 3.4.5 , "RCS Leakage Detection Instrumentation". Your letter documented information previously discussed with the NRC in a telephone conference on May 12, 2005 at 1050 EDT. The principal NRC staff members who participated in that telephone conference in[illegible]d from Region II, Charles Casto, Director, Division of Reactor Safety, Anne Boland, Acting Deputy, Division of Re[illegible] Projects (DRP), and Paul Fredrickson, Chief, DRP Branch 4, and from the Office of Nuclear Reactor Regulation, Edwin Hackett, Director, Project Directorate II, Herbert Berkow, Project Director, Project Directorate IV, Michael Marshall, Section Chief, Section II-2, Project directorate II and Brenda Mozafari, Sr. Project Manager Brunswick Steam Electric Plant, Project Directorate II. You stated that on May 12, 2005, at 1111, Unit 1 would not be in compliance with LCO 3.0.3., which required the unit be in Mode 2 (Startup/Hot Standby) 7 hours after the Unit failed to meet LCO 3.4.5 at 0411 on May 12. You requested that a Notice of Enforcement Discretion (NOED) be granted pursuant to the NRC's policy regarding exercise of discretion for an operating facility, set out in Section VII.C, of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, and be effective for the period from 1111 on May 12, to 0011 on May 13. This letter documents our telephone conversation on May 12 at 1050, when we orally issued this NOED. We understand that the condition causing the need for this NOED was corrected by you causing Unit 1 to exit from LCOs 3.0.3 and 3.4.5 and from this NOED on May 12 at 1740.

At 0411 on May 12, the feeder breaker to emergency bus E1 tripped. Additionally, emergency diesel generator (EDG) 1, which provides emergency power to bus E1, had been taken out of service on May 9, for routine maintenance and surveillance testing. As a result, there was no power to bus E1. Without bus E1, the drywell floor drain sump flow monitoring system, and the primary containment atmosphere particulate and the primary containment atmosphere gaseous RCS leakage monitoring systems were rendered inoperable. This was primarily the result of closure of the containment isolation valves associated with the systems and the inability to re-open the valves due to the inoperable bus E1. With all required leakage detection systems inoperable, Unit 1 entered Condition D of LCO 3.4.5 at 0411 on May 12. Condition D required immediate entry into LCO 3.0.3. Compliance with LCO 3.0.3, required Unit 1 to be in Mode 2 by 1111 on May 12. We understand from your letter that at 0948, a Unit 1 power reduction, in accordance with LCO 3.0.3 was initiated. However, loss of bus E1 also affected the ability to reduce power using the 1A reactor recirculation (RCR) pump, with the 1A MG set scoop tube being locked due to the loss of control power. In order to achieve Mode 2 by 1111, power needed to be reduced via the 1B RCR pump until limited by RCR pump mismatch requirements at which time power reduction via co[illegible]rod insertion continued. The time required to either pre-brief and safely perform manual operation of the scoop tube or [illegible]nue to reduce power via control rod insertion alone, did not support entry into Mode 2 by 1111. Therefore, operations intended to insert a manual scram at 1100 to comply with LCO 3.4.5. It was estimated that Unit 1 would be at approximately 60 percent of rated thermal power at the time the manual scram was required.

Your May 13 letter stated that the proposed NOED would avoid an unnecessary transient as a result of compliance with LCO 3.4.5. If compliance with LCO 3.4.5. was enforced, a relatively high powered unnecessary manual scram (i.e., approximately 60 percent of rated thermal power), complicated by the inoperable bus E1, would have been required. Your letter also stated that the requested NOED would also allow for necessary planning and precautions to be implemented, as appropriate, to conduct an orderly shutdown, given the degraded condition of the electrical distribution system. In the interim, as a compensatory measure, Operations would perform increased monitoring of the narrow range primary containment pressure instrumentation and the primary containment temperature instrumentation—data to be logged every 30 minutes. The drywell temperature and pressure provide the indirect ability to detect increased RCS leakage. Your letter concluded that the proposed NOED would not be of potential detriment to the public health and safety, because it provided additional time to plan and implement, if necessary, a more orderly shutdown of Unit 1. This time was consistent with existing shutdown requirements in TS 3.4.4, for actual RCS leakage conditions in excess of TS requirements, and in TS 3.8.7, for degraded electrical system conditions. The letter also stated that there was no net increase in radiological risk to the public by avoiding the unnecessary transient imposed through compliance with LCO 3.0.3 and safety continued to be assured by the existing LCO requirements of TS 3.8.7 which included more appropriate completion times when dealing with a degraded electrical distribution system.

The resident inspectors reviewed the difference in plant response from scrams from high power and lower power. Lower power scrams would result in smaller transient response and, therefore, were less likely to challenge safety systems than high power scrams. The residents also confirmed that with the additional time, the licensee would be better prepared for any problems that might occur from a scram (at any power level) coincident with having an unpowered emergency bus. Based on discussion with the licensee and the reviews conducted by the resident inspectors, the staff determined the request was acceptable.

In the May 13 letter, you committed to submit a follow-up amendment request by May 17 , addressing the existing Condition D of LCO 3.4.5. Also, based on our May 12 telephone conference, we understand that this amendment request will permanently align the allowed outage time for the loss of all required leak detection systems closer to that specified in LCO 3.8.7 for the loss of an AC electrical power distribution subsystem, which initiated this loss of all leak detection. We plan to complete our review and disposition of your follow-up license amendment request within 4 weeks of the date of this letter.

On the basis of the staff's evaluation of your request, we have concluded that granting this NOED is consistent with the Enforcement Policy and staff guidance, and has no adverse impact on public health and safety or the environment. Therefore, it is our intention to exercise discretion to not enforce compliance with LCOs 3.0.3 and 3.4.5 for the period from 1111 to 1740 May 12 . Your need for the NOED differed from the approved NOED effective duration because the sample systems were returned to service prior to 0011, May 13.

As stated in the Enforcement Policy, action will be taken, to the extent that violations were involved, for the root cause that led to the noncompliance for which this NOED was necessary.

Sincerely,

**/RA/**

Loren R. Plisco
Deputy Regional Administrator

Docket No.: 50-325

cc:

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Progress Energy Carolinas, Inc.
Electronic Mail Distribution

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Assistant Attorney General
State of North Carolina
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*Last revised Tuesday, May 24, 2005*

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

May 6, 2005

NRC INFORMATION NOTICE 2005-11: INTERNAL FLOODING/SPRAY-DOWN OF SAFETY-RELATED EQUIPMENT DUE TO UNSEALED EQUIPMENT HATCH FLOOR PLUGS AND/OR BLOCKED FLOOR DRAINS

**ADDRESSEES**

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

**PURPOSE**

The U.S. Nuclear Regulatory commission (NRC) is issuing this information notice (IN) to inform addressees of the possibility of flooding safety-related equipment as a result of (1) equipment hatch floor plugs that are not water tight and (2) blockage of the equipment floor drain systems that are credited to mitigate the effects of flooding in the final safety analysis report (FSAR) and plant design basis calculations.

It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions in this IN are not NRC requirements; therefore, no specific action or written response is required.

**DESCRIPTION OF CIRCUMSTANCES**

On August 18, 2004, during an extended backwash evolution on the reactor water cleanup demineralizer at Susquehanna Unit 1, approximately 1,500 gallons of contaminated water from the backwash-receiving tank overflowed into the reactor building equipment floor drain system. The drain header became blocked by the resin from the receiving tank and rust displaced from inside the drain pipes. The water flowed up and out of the blocked drains on a lower elevation, across the floor, and down into the Division II core spray and the high-pressure coolant injection system compartments. The water entered these compartments by flowing though unsealed spaces between the equipment hatch floor plugs and the floor. The equipment floor drains in the emergency core cooling system (ECCS) compartments were isolated, as per design, and the water accumulated on the floor. Approximately 2 inches of water accumulated on the floor of the Division II core spray and high-pressure injection system compartments.

**ML050870351**

The licensee disabled the automatic start feature of the Division II core spray pumps and declared the pumps inoperable but functional for approximately 2hours until an assessment was completed to verify operability. The licensee's immediate actions were to perform system walkdowns and determine the effect of the flooding on the Division II core spray and high-pressure coolant injection systems. The walkdowns determined that the water leakage into the room did not significantly impact the systems and that the systems remained operable. The walkdowns also confirmed that the other ECCS compartments were not affected.

## DISCUSSION

The licensee's FSAR flooding analysis stated that the ECCS compartments were water tight. However, the equipment hatch floor plugs were not sealed and therefore did not constitute a water-tight barrier. The plant-specific design calculations did not address leakage around these plugs. Water leaked through the ceiling into the Division II core spray and high-pressure coolant injection compartments through gaps between the equipment hatch floor plug and the floor. The licensee did not scope the equipment floor drain system function (mitigating internal flooding) into the Maintenance Rule. The FSAR and flooding design calculations credited the equipment floor drain system to assist in removal of water from internally flooded areas. Part of the licensee's corrective actions were to place the floor and equipment drain functions within the scope of the Maintenance Rule and to revise the plant calculations to reconcile the plant design analysis and the FSAR. The licensee has also developed and installed a modification to seal the equipment hatch floor plugs above the ECCS compartments.

## CONCLUSION

The event described above illustrates the potential for degradation of multiple trains of ECCS as a consequence of potential flooding of safety-related equipment outside containment. All holders of reactor operating licenses should be aware of the effect of unsealed equipment hatch floor plugs and malfunctioning drains on their plant-specific FSARs and plant design analyses for flooding. Unsealed reactor building equipment hatch floor plugs and less than optimal maintenance monitoring of the equipment floor drain systems can result in additional internal flooding vulnerabilities. Licensees should consider several actions to mitigate these vulnerabilities:

(1) Verify that the actual plant configuration and design of the equipment hatch floor plugs meet the FSAR description and design basis calculations for water-tight compartments. Consider appropriate actions to achieve floor plug water tightness, such as the use of gaskets and sealers. Additionally, if the analysis allows for leakage through the equipment hatch floor plugs, the qualification of equipment below these plugs should consider water spray as well as submergence.

(2) In addition, if the drain system is required for water removal, periodically verify that the floor drain system performs as intended and perform maintenance to assure that the system can perform the water removal function assumed in the FSAR and design calculations.

## GENERIC IMPLICATIONS

Flooding due to internal causes has been shown to be a significant contributor to risk at some facilities. Flooding also has the potential to make multiple trains of equipment and support equipment inoperable, significantly increasing plant risk. Flooding also has the significant consequence of preventing or limiting operator mitigation and recovery actions. As a result, semiannually, NRC inspectors select one or two plant areas and inspect internal flood protection features for risk-significant structures, systems, and components in accordance with Inspection Procedure Attachment 71111.06 "Flood Protection Measures."

Multiple previous NRC generic communications have addressed flood protection issues:

1. Circular CR 78-06 — POTENTIAL COMMON MODE FLOODING OF ECCS EQUIPMENT ROOMS AT BWR FACILITIES
2. IN 83-44 — POTENTIAL DAMAGE TO REDUNDANT SAFETY EQUIPMENT AS A RESULT OF BACKFLOW THROUGH THE EQUIPMENT AND FLOOR DRAIN SYSTEM
3. IN 83-44 S1 — POTENTIAL DAMAGE TO REDUNDANT SAFETY EQUIPMENT AS A RESULT OF BACKFLOW THROUGH THE EQUIPMENT AND FLOOR DRAIN SYSTEM
4. IN 87-49 — DEFICIENCIES IN OUTSIDE CONTAINMENT FLOODING PROTECTION
5. IN 92-69 — WATER LEAKAGE FROM YARD AREA THROUGH CONDUITS INTO BUILDINGS
6. IN 94-27 — FACILITY OPERATING CONCERNS RESULTING FROM LOCAL AREA FLOODING
7. IN 98-31 — FIRE PROTECTION SYSTEM DESIGN DEFICIENCIES AND COMMON-MODE FLOODING OF EMERGENCY CORE COOLING SYSTEM ROOMS AT WASHINGTON NUCLEAR PROJECT UNIT 2

**CONTACTS**

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contact(s) listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

April 19, 2005

# NRC REGULATORY ISSUE SUMMARY 2005-07 COMPENSATORY MEASURES TO SATISFY THE FIRE PROTECTION PROGRAM REQUIREMENTS

## ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

## INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform addressees that alternate compensatory measures as otherwise required by the approved fire protection program may be used for a degraded or inoperable fire protection feature under certain circumstances, in accordance with the discussion in this RIS. This RIS describes the proper method for changing the approved fire protection program (FPP) to use an alternate compensatory measure. No action or written response is required.

## BACKGROUND INFORMATION

The purpose of this RIS is to discuss how a licensee, with the standard license condition for fire protection, may change the approved FPP to use alternate compensatory measures. This RIS is in response to industry feedback from the October 13-15, 2004, public workshop in Atlanta, GA (ADAMS Accession No. ML043430240) on implementing changes to compensatory measures in the approved FPP. Generic Letter 86-10, "Implementation of Fire Protection Requirements," dated April 24, 1986, specified a process for revising the operating license condition to allow licensees to remove fire protection technical specifications and put them in the approved FPP. The standard license condition for fire protection is given in Generic Letter 86-10:

> Fire Protection
>
> (Name of Licensee) shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility (or as described in submittals dated _____) and as approved in the SER dated _____(and Supplements dated _____) subject to the following provision: The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

**ML042360547**

Currently most licensees have adopted this license condition and moved the fire protection compensatory measures from technical specifications to the approved FPP in accordance with the guidance in Generic Letter 88-12, "Removal of Fire Protection Requirements From Technical Specifications," dated August 2, 1988.[1]

**SUMMARY OF THE ISSUE**

In certain situations a licensee may prefer to implement an alternate compensatory measure to the one required by the FPP. For example, the compensatory measure required by a licensee's approved FPP for a degraded fire barrier is typically an hourly fire watch or, in the case of an inoperable fire suppression system, a continuous fire watch with backup fire suppression equipment. Fire watches may not be the most effective compensatory measure for degraded or inoperable fire protection features or post-fire safe-shutdown capability (see Information Notice 97-48). A licensee may choose to implement a different compensatory measure or combination of measures (e.g., additional administrative controls, operator briefings, temporary procedures, interim shutdown strategies, operator manual actions, temporary fire barriers, temporary detection or suppression systems). Such a change may be made to the approved FPP. However, the licensee must perform a documented evaluation of the impact of the proposed alternate compensatory measure to the FPP and its adequacy compared to the compensatory measure required by the FPP. The evaluation must demonstrate that the alternate compensatory measure would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Any change to the FPP must maintain compliance with the General Design Criteria and 10 CFR 50.48(a), and must be retained as a record pursuant to 10 CFR 50.48(a). The licensee's change to the FPP is subject to inspection by the NRC through the Reactor Oversight Process (ROP).

The evaluation of the alternate compensatory measure should incorporate risk insights regarding the location, quantity, and type of combustible material in the fire area; the presence of ignition sources and their likelihood of occurrence; the automatic fire suppression and fire detection capability in the fire area; the manual fire suppression capability in the fire area; and the human error probability where applicable.

As a reminder to licensees, additional information on compensatory measures can be found in Information Notice 97-48, "Inadequate or Inappropriate Interim Fire Protection Compensatory Measures," dated July 9, 1997. Information Notice 97-48 provides background on the use of compensatory measures, discusses situations in which fire watches were used inappropriately, and gives examples of alternate compensatory measures that may be preferred for degraded or inoperable conditions associated with post-fire safe-shutdown capability or degraded fire protection features. Another source of information is Generic Letter 91-18, Revision 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," dated October 8, 1997, which discusses, among

[1] Licensees without the standard license condition for fire protection may have to submit changes to their FPP for NRC review and approval. Licensees may not change a compensatory measure required by the licensee's technical specifications without NRC review and approval by license amendment.

other topics, compensatory measures and timely corrective actions.[2] Licensees may want to review and consider guidance provided to NRC inspectors and discussed in the most up-to-date generic communication on timeliness of corrective actions. The reliance on compensatory measures is one of the factors in the inspector guidance when evaluating timeliness of the corrective action. The expectation in the guidance is to promptly complete the corrective action at the first available opportunity and eliminate the reliance on the compensatory measure.

## BACKFIT DISCUSSION

This RIS requires no action or written response and is, therefore, not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis.

## *FEDERAL REGISTER* NOTIFICATION

A notice of opportunity for public comment was not published in the *Federal Register* because this RIS is informational.

## SMALL BUSINESS REGULATORY ENFORCEMENT FAIRNESS ACT OF 1996

The NRC has determined that this action is not a rule and thus is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

## PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain information collections and, therefore, is not subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.).

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[2] As of the date of this RIS, the most current guidance is Generic Letter 91-18, Revision 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," dated October 8, 1997, ADAMS Accession No. ML031200701 and Attachment 1 to Generic Letter 91-18, dated October 8, 1997, ADAMS Accession No. ML031200704.

## CONTACT

Please direct any questions about this matter to the technical contacts listed below or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

# NRC NEWS

**U. S. NUCLEAR REGULATORY COMMISSION**
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## NRC TO CONDUCT SPECIAL INSPECTION AT WATERFORD NUCLEAR PLANT

The Nuclear Regulatory Commission will conduct a special inspection into unexpected conditions that developed while operators were preparing the reactor at the Waterford 3 nuclear plant for refueling. Entergy Operations runs the plant, located near Taft, La.

On April 19, operators were lowering the level in the reactor coolant system in preparation for lifting the reactor vessel head to begin refueling. Because operators failed to open valves for air to enter the reactor coolant system as the water level was lowered, vapor bubbles formed. The bubbles were pushed through the shutdown cooling system, causing flow to fluctuate.

When operators attempted to open an access hatch in the pressurizer, they subsequently discovered that a vacuum had formed in the reactor coolant system. The vacuum condition was then corrected. Although there was no immediate safety concern, following analysis, the NRC staff has decided to examine the circumstances leading up to the event in a more detailed way using a special inspection.

"While public health and safety were not impacted by this event, we have questions about operator performance and plant configuration control that need to be answered," Region IV Administrator Bruce S. Mallet said. "If there are lessons to be learned from this event, we will share them with other nuclear plant operators."

The NRC will dispatch a three-person special inspection team to the site in June to better understand the circumstances associated with the event, the licensee's response to it, and its risk significance. Prior to the team's arrival on site, resident inspectors are continuing to monitor the licensee's ongoing review of the event.

The NRC inspection team includes specialists from the Region IV office in Arlington, Texas. The team is expected to spend about one week on site before returning to the regional office, where information they gather will be analyzed and evaluated.

The report of the special inspection will be publicly available when it is issued, about 30 days after the close of the special inspection. It will be posted in the NRC's electronic reading room at: http://www.nrc.gov/reading-rm/adams/web-based.html . To locate the report, once it is issued, enter the docket number for the Waterford plant (50-382) in the search phrase box.

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# NRC NEWS

**U.S. NUCLEAR REGULATORY COMMISSION**

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No. 05-077 May 5, 2005

## NRC ANNOUNCES NEW DIRECTOR OF ENFORCEMENT

The Nuclear Regulatory Commission is announcing the appointment of Michael R. Johnson as director of the agency's Office of Enforcement, effective May 30, 2005. In this position, he will be responsible for managing the development and implementation of programs that enforce NRC requirements. He will also oversee the agency's external allegations review process and internal Differing Professional Opinions Program.

Johnson joined the NRC in 1986 as a reactor operations engineer. Later, he served as regional coordinator in the Office of the Executive Director. Since 1989, he has served in a number of increasingly responsible posts within the agency's Office of Nuclear Reactor Regulation, including leadership positions in the Inspection Program and Assessment Section, Performance Assessment Section, Inspection Program Branch and the Probabilistic Safety Assessment Branch. Most recently, he held the title of deputy director of the Division of Systems Safety and Analysis.

Johnson is a graduate of the NRC's Senior Executive Service Candidate Development Program. He received a bachelor of science degree in Ocean Engineering from the U.S. Naval Academy. He resides in Silver Spring, Md.

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# NRC NEWS

**U.S. NUCLEAR REGULATORY COMMISSION**
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## NRC TERMINATES SPECIAL OVERSIGHT PANEL, BEGINS AUGMENTED REACTOR OVERSIGHT PROGRAM FOR DAVIS-BESSE

The Nuclear Regulatory Commission staff has terminated its special Oversight Panel for the Davis-Besse Nuclear Power Plant and returned the monitoring of the facility to an expanded version of the agency's Reactor Oversight Process.

The agency established its Davis-Besse Oversight Panel in April 2002 to coordinate the NRC's regulatory activities in response to the corrosion damage to the reactor vessel head and other safety issues at the plant. The panel included NRC managers and staff from the Region III office in Lisle, Ill., NRC Headquarters in Rockville, Md., and the NRC resident inspectors at the plant.

Davis-Besse -- operated by FirstEnergy Nuclear Operating Company and located near Oak Harbor, Ohio – resumed operation in March 2004 after being shut down for about two years to address equipment and staff performance problems.

"Since Davis-Besse returned to operation, the plant has operated safely," said James Caldwell, NRC Regional Administrator. "With the record of safe operation at power for more than a year now and their satisfactory completion of the order requiring independent assessments and a successful mid-cycle outage, the NRC can make the transition from a special panel to an augmented oversight program."

The NRC Oversight Panel will meet with FirstEnergy officials on May 24 to discuss the transition. The meeting, which will be open to public observation, will begin at 6 p.m. at the Camp Perry Clubhouse, 1000 Lawrence Road, Bldg. 600, in Port Clinton. At the conclusion of the meeting, before adjournment, the NRC staff will be available for public questions and comments.

The Reactor Oversight Process includes monitoring of plant safety performance with statistical measures called performance indicators and inspections of plant safety performance and activities by NRC resident inspectors and specialists based in the regional office and NRC headquarters.

Although oversight of the plant will be returned to the normal program, it will be substantially expanded to include additional inspections focused on corrective action program implementation and additional requirements imposed on the plant in an NRC Confirmatory Order issued at the time of startup.

The Confirmatory Order required Davis-Besse to contract with outside organizations to perform independent assessments in the areas of operations, engineering, corrective actions, and safety culture. In addition, FirstEnergy established action plans to reduce the backlog of maintenance and corrective action items, improve the corrective action program, and ensure a safety culture where individuals feel free to raise safety issues.

The NRC will perform inspections beyond the regular oversight program to monitor these independent assessments and the status of the action plans.

The NRC uses a color-coded system to measure the safety significance of problems and inspection findings at nuclear plants ranging from green, through white and yellow, to red, the most significant.

A white finding at Davis-Besse dealing with inoperable emergency sirens identified during testing in May 2004, remains open and will result in an additional inspection later this year to review the utility's corrective actions.

The letter notifying FirstEnergy of the transition from the NRC Oversight Panel to the normal Reactor Oversight Process is available on the NRC's web site at: http://www.nrc.gov/reading-rm/adams/web-based.html. Use accession number ML051400049 to locate the document.

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# NRC NEWS

**U.S. NUCLEAR REGULATORY COMMISSION**

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## NRC REPORTS PRELIMINARY RESULTS OF MILLSTONE 3 SPECIAL INSPECTION

An NRC special inspection team sent to the Millstone 3 nuclear power plant to review an April 17th shutdown has preliminarily concluded the event was caused by an apparent failure of a circuit card in a computerized reactor protection system. The team has also determined the unit was safely removed from service by operators, any radiological releases were well below regulatory limits and there were no public health and safety consequences. However, the team did identify several inspection findings.

NRC staff will present the preliminary results of the inspection this evening at a meeting of Connecticut's Nuclear Energy Advisory Council (NEAC). NEAC's meeting, which is open to the public for observation, is scheduled to get under way at 6 p.m. at Waterford Town Hall, 15 Rope Ferry Road in Waterford, Conn.

The Millstone 2 and 3 nuclear power plants are located in Waterford and operated by Dominion Resources.

On April 17, Millstone 3 experienced an automatic shutdown from full power. The event involved the activation of various safety subsystems. As the reactor was shutting down, main steam line safety valves opened to automatically remove heat generated after the reactor shutdown. One main steam line safety valve remained open for an extended period of time. It was this condition that resulted in the declaration of an "Alert" - the second-lowest of four levels of emergency classification.

In response to the event, the NRC sent a special inspection team to the facility on April 20 to evaluate the circumstances surrounding the shutdown with complications. Consisting of four full-time and four part-time inspectors, the team was a multi-disciplined group with diverse backgrounds.

The team has identified several findings, including issues involving the failure of a turbine-driven pump that is part of a system sometimes used to help cool down the plant, control-room indicator design issues, minor operator response issues, a delay in the activation of a computer program used to evaluate plant conditions and the adequacy of the company's boric acid control program. The NRC has preliminarily determined these findings to be "green."

The NRC uses color-coded inspection findings and performance indicators to assess nuclear power plant performance. The colors start with "green" and then increase to "white," "yellow" or "red," commensurate with the safety significance of the issues involved.

The special inspection team has also noted that in response to the event, Dominion made a timely assessment of radiological

conditions and has since launched a review of the shutdown by its own evaluation team, implemented multiple corrective actions and planned additional evaluations.

After presenting its preliminary findings this evening, the NRC special inspection team will document its findings and conclusions in a report to be issued within 45 days.

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*Last revised Thursday, May 19, 2005*

Inside NRC
Volume 27 / Number 10 / May 16, 2005

## Performance indicators in oversight program could use fix, staff says

A backlog of unresolved performance indicators (PIs) and concern that some PIs are not signaling waning plant performance have raised questions about the effectiveness of that part of NRC's reactor oversight process (ROP), staff said in a paper released last week. Several lawmakers also recently questioned how NRC's oversight program is faring. Two senators and three congressmen have requested the Government Accountability Office (GAO), Congress' investigative unit, to review the effectiveness of the ROP, with particular attention to how the process tracked performance at Hope Creek.

Sens. Joseph Biden (D-Del.) and Tom Carper (D-Del.) and Reps. Michael Castle (R-Del.), Ed Whitfield (R-Ky.), and Edward Markey (D-Mass.) acknowledged in an April 14 letter to the GAO that there have been relatively few plants with either overall safety significant performance issues or multiple or repetitive degraded conditions. But they said they wanted an independent review of NRC's oversight process and of related issues, such as whether the NRC is taking appropriate enforcement action.

Meanwhile, the staff said it will work with senior industry leaders to address what it sees as shortcomings with some of the Pis.

As it has done for the past four years, the staff has sought input from licensees on the agency's regulatory performance and incorporated those responses into a selfassessment.

A large majority (85%) of the comments were favorable, with most of the unfavorable responses (8%) relating to the agency's handling of security and safeguards matters. Only 5% provided feedback on the ROP, with about half finding problems in particular areas of the program, such as the significant determination process (SDP), which characterizes the safety significance of inspection findings. The staff itself said it believed the ROP had in general met its program and performance goals, including being objective, understandable and predictable, maintaining safety, enhancing public confidence, reducing unnecessary regulatory burden, and making activities more effective and efficient.

But certain elements of the ROP need improvement, the staff said in its paper, Secy 05-70.
The PI program has had success in focusing industry attention on certain areas and improving those activities, such as emergency drills/exercises, the staff said.

**Unchanging green**

But some PIs have always remained "green," or within expected performance limits. The staff said it is concerned that those PIs provide little insight into plant performance and should be re-examined. The "number of PIs that are consistently green at all plants bring into question the present value of those PIs to the program," the staff said.
When the ROP was developed, it was anticipated that about 5% of the plants would exceed the green-white threshold for each PI, the staff said. But in practice, only 0.60% of the PIs have been reported as greater than green since the ROP was implemented in 2000. Last year only 0.47% of reported PIs were greater than green, the staff said.
Performance indicators are combined with inspection findings into an "action matrix" to determine a plant's margin of safety and appropriate level of NRC attention. PIs that fall into the green band of the ROP, which has the lowest safety significance in the four-tier system, need only baseline inspection by NRC. A white PI might trigger additional oversight, but a yellow or red PI automatically increases NRC oversight.

The staff looked at six cases in which reactors moved under the oversight of a special NRC panel or into so-called Column 4 of the action matrix—the multiple/degraded cornerstone column—which indicates a plant has a significant reduction in safety margin. In each instance, the PIs did not flag the major problem that resulted in the plant entering Column 4 or getting into an additional oversight situation.
"Based on the historical results, the staff has concluded

that the PI program has not contributed to the early identification of poor performing sites to the degree envisioned by the staff," it said.

But the staff said it believes two PIs—unplanned power changes and safety system functional failures—to be "leading indicators," meaning predictors of performance. The staff said a third PI—scrams with loss of normal heat removal—has the "potential to be a leading indicator of declining performance." Most of the backlog for the frequently-asked-question (FAQ) process deals with issues relating to the scrams with loss of normal heat removal.

The FAQ process was established to handle licensee questions about the PIs. These issues are worked out through a joint NRC-industry working group, and until last year, there was no process for settling disputes. Now when a consensus can't be reached, the issue is brought before the director of NRC's division of inspection program management. But the staff said that the industry "does not view the process as retroactive, so there remain a number of old open questions that are yet to be resolved."

The staff said that one problem with the FAQ process is that many of the issues have taken too long to resolve. There are about 11 FAQs in the backlog on the scram with loss of heat removal PI. "Several of them are more than three years old, which means they no longer contribute to the PI calculation, and hence the color of the PI," the staff said, noting that this particular PI counts such scrams over the previous 12 quarters.

Anthony Pietrangelo, senior director of risk regulation for the Nuclear Energy Institute (NEI), said he doesn't believe the staff has highlighted any new issues. "We're working on the PIs," he said.

Since the inception of the ROP, the NRC-industry task force has closed about 370 issues, Pietrangelo said. Between three and five FAQs are raised each month and are resolved in an average of three months, he said, adding that there are roughly between 10 and 15 FAQs in the backlog at any given time. A meeting last week that focused on unresolved scrams with loss of normal heat removal PIs could result in the disposition of about half of the backlog, Pietrangelo said.

Pietrangelo noted that the PIs were only part of the oversight program. "PIs aren't supposed to be a catchall," he said. "They complement [NRC's] baseline inspection

program....Why aren't they doing the same analysis on inspection findings as they did on Pis?"

### Inspection program

The number of inspection findings increased in 2004 compared to the previous year, rising to 893 findings from 733. Twelve inspections findings were characterized as white, while the remainder were found to be green. There were no yellow or red findings last year. The staff said in its paper that the increase in findings could be an indicator of a declining performance trend in the industry, although it doesn't have data yet to support that possibility.
More likely, the staff said, the increase in findings was tied to either an increase in events or the staff's attention to plants operating in Column 4.

NRC staff reported that inspection expenditures have decreased "significantly" over the past 10 years but that savings are now leveling off "as available efficiencies are exhausted." The staff said there might be a few other ways it could cut program costs, such as by implementing the revised design engineering inspections that have been piloted, providing credit for licensee self-assessments, and making timeliness improvements to the SDP.

### MSPI launch

Other savings to the inspection program could come through implementation of the mitigating system performance index (MSPI), which is slated to replace the existing safety system unavailability PI in January. The staff said in its paper that the launch could be impacted by probabilistic risk assessment (PRA) quality issues. The staff said that the industry has recently indicated that there are "open PRA issues relevant to MSPI at an estimated 50 percent of the plants, with some facilities having as many as 30 or 40 issues to resolve."

The staff said NEI has the "lead responsibility" to prepare the operating reactors for the MSPI implementation. Pietrangelo said last week that NEI and the industry are "working to make it happen."—***Jenny Weil, Washington***

Inside NRC
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## Entergy makes pitch to NRC staff to approve Vermont Yankee uprate

Entergy managers said last week they have completed all the engineering necessary for NRC staff to make a decision on the company's 20% uprate request for Vermont Yankee and that the plant is technically capable of handling the additional power generation.

The May 9 meeting, held at Entergy's request, was intended to provide an overview of the company's latest filings responding to staff questions. But Entergy managers also stressed the importance of reaching a decision soon. NRC's original schedule targeted a decision in January. But questions about the performance of the unit's steam dryer under uprated conditions lengthened the review. NRC staff remained non-committal about a new schedule last week, saying it has to review the new information Entergy submitted.

The company wants to implement the uprate by early fall so the steam dryer will have had time to run under the higher operating level before the unit has to refuel in September. The company is pressing NRC because it wants to be able to inspect the steam dryer when it shuts down.

"If it means camping out on your doorstep, that's what we'll do," said one Entergy manager. "It's that important to us."

Vermont Yankee made many uprate modifications during an April 2004 outage. The company says it is ready for 115% operation and will complete the work necessary to reach the full 120% uprate in the fall. Those modifications relate to ensuring the stability of the grid, said John Dreyfuss, Vermont Yankee's director of engineering.

Most of the two-hour meeting last week was devoted to Vermont Yankee's steam dryer analysis. The steam dryer is "one of the most challenging aspects of the uprate," said Brian Hobbs, Vermont Yankee project engineer. Entergy has proposed a license condition outlining a "slow and deliberate" power increase, with frequent data collection and evaluation points, he said. "The bottom line," he said, "is Vermont Yankee won't generate loose parts operating in the uprate condition."

NRC began focusing on steam dryers after cracks were found at four Exelon units—Quad Cities-1 and -2 and Dresden-2 and -3—that had been operating under extended power uprates. EPUs are considered thermal power increases above 7%. NRC is concerned that loose parts from damaged steam dryers, which are non-safety-related equipment, could migrate and cause harm to safety-related components (INRC, 4 Oct. '04, 10).

Hobbs said in addition to extensive modeling that Vermont Yankee conducted, it also has made significant modifications to strengthen the steam dryer. It has attached eight strain gages on the main steam lines to measure flow induced vibration, installed four pressure transmitters on the main steam venturis, and put 31 accelerometers in the drywell. Based on the instrumentation measurements, Entergy has concluded that vibration is "very low," Hobbs said. He said Entergy installed gussets last year on the steam dryer, something that has not been added to Monticello, which has the same dryer design and has been operating for the past six years at an EPU level.

Entergy says it has answered 161 requests for additional information (RAI). But Raymond Shadis, a technical adviser to the advocacy group New England Coalition, said at last week's meeting that Vermont Yankee's answers may not have been "fully responsive" to the RAIs. He said some responses reviewed by NEC contained "subjective generalities" or lacked a rationale for the conclusions provided. For example, Shadis noted one Vermont Yankee response in an April 22 filing to NRC in which Entergy said the "CP [condensate pump] flow margin between the EPU conditions and the runout with three RFPs [reactor feedwater pumps] and three CPs will be approximately 7% greater than the required EPU flow. Industry criteria typically recommend a 5% margin. As such, the available margin exceeds that typically required by the industry."

Shadis said he was unsure what was meant by "typically required by the industry" and whether that margin is really adequate or not. "On the whole," Shadis said, "the RAIs were not responded to in a forthcoming way."

NEC is challenging Vermont Yankee's uprate request before an NRC judicial panel. Shadis told the staff that its review schedule for the application would have an impact on that proceeding as well. The Atomic Safety & Licensing Board's schedule for completing the hearing is tied to the staff's completion of a safety evaluation report.
***—Jenny Weil, Washington***

Inside NRC
Volume 27 / Number 10 / May 16, 2005

## IRSN inaugurates facility to test iodine in severe accidents

France's Institute of Radiological Protection & Nuclear Safety (IRSN) will inaugurate next week a new facility at Cadarache destined to study the behavior of radioactive iodine during a severe accident in a PWR.

The Epicur test rig is part of IRSN's "source term" research program, which is to run between 2005 and 2010 and is complementary to the much costlier programs the institute has run at the Phebus experimental reactor nearby. Michel Schwarz, director of IRSN's Severe Accident Prevention Division, said Source Term would be co-financed by a tripartite French team of IRSN, the Commissariat a l'Energie Atomique (CEA), and Electricite de France. In addition, he said, IRSN is in the final stages of negotiating agreements for participation with the NRC, as well as the European Commission, Switzerland's Paul Scherrer Institut, and the Korea Atomic Energy Research Institute. Other international organizations are interested in participating, but discussions are less advanced, Schwarz said May 12.

Phebus, which is designed to reproduce severe accidents in a test reactor representative of a PWR primary circuit, had its last "Fission Products" series test last fall. IRSN and potential partners are exploring the potential for refitting the reactor to study other aspects of severe accidents, in particular loss-of-coolant accidents (LOCA). The so-called ST LOC (for "source term" and "LOCA") test series has been estimated to cost about Eur 150-million (U.S.$190-million) and is now proposed to begin after 2010 (INRC, 18 Oct.'04, 8). An international experts' group is to submit conclusions on Phebus' future role by the end of 2006 (INRC, 29 Nov. '04, 6).

Schwarz said the Source Term program would cost about Eur 30-million, including separate experiments in Epicur to study the behavior of ruthenium in the presence of oxygen, for example in the event of air ingress into the reactor vessel, and a separate series of experiments on fission product release from irradiated fuel at a CEA facility called Verdon. Experiments to study iodine chemistry will also be run in another new facility at Cadarache, called CHIP.

Epicur is basically an irradiator that reproduces flux conditions of a power reactor during a core-melt accident, with a dose rate of 10 Gray per hour. An iodine solution is placed inside the irradiation chamber, with an iodine-131 tracer, allowing study of how organic iodine is formed and destroyed during such an accident.

The idea behind the Epicur and CHIP tests, Schwarz said, is to allow precise estimation of how much organic iodine would be formed and could be released in a severe accident. Previous Phebus tests have shown that a greater than expected quantity of organic iodine was formed during a core melt event. The sand filters at French nuclear power plants, which are connected to the containment by a dedicated pipe, are designed for controlled containment venting to avert dangerous overpressure. They retain aerosols and molecular iodine, but not organic iodine, Schwarz noted. He added that organic iodine would also penetrate containment basemats more easily than aerosols or molecular iodine.

Schwarz said researchers postulate that organic iodine found in the Phebus circuit may have been created by a chemical imbalance as elements cooled after the core melt. He said it's thought that silver-indium-cadmium control rods played a role in these chemical reactions, but not known which element was primarily responsible. The upcoming tests, he said, were designed to "isolate the essential element that explains" the phenomenon observed in Phebus.

He said the NRC is also particularly interested in the planned Epicur studies of highly toxic radioactive ruthenium which could be released if oxygen penetrated the reactor vessel, in the so-called chimney or barbecue effect. Schwarz said the first Epicur test was being conducted last week.—*Ann MacLachlan, Paris*

## *NUCLEAR NEWS FLASHES - Thursday, May 26, 2005*

**U.S. NEWS: --INVESTMENT INCENTIVES FOR NEW NUCLEAR POWER PLANTS WERE INCLUDED** in comprehensive energy legislation that was approved by the Senate Energy & Natural Resources Committee today. The incentives are for new or advanced energy technologies that avoid or reduce air pollutants or greenhouse gases. The loan guarantee program would encourage investments in new nuclear plants, the Nuclear Energy Institute (NEI) said. "Limited federal investment for a limited number of new nuclear plants for a limited period of time will jump-start the construction of the first advanced-design nuclear power plants as they navigate" the NRC's new licensing process, said John Kane, NEI's senior vice president of governmental affairs. The bill was approved after senators from both sides of the aisle decided to wait to submit controversial changes, such as proposals to encourage new offshore drilling, until the legislation reaches the full Senate, which is expected to consider it next month.

## *NUCLEAR NEWS FLASHES - Thursday, May 26, 2005*

**--NUCLEAR ENERGY IS AT A CROSSROADS, WITH THE NEXT FEW MONTHS** and years determining whether the technology will thrive in the 21st century, Sen. James Inhofe (R-Okla.), chairman of the Environment & Public Works Committee, said today. Calling nuclear power safe, low-cost, and environmentally friendly, Inhofe said "there is no reason why in the long term, nuclear power can't or shouldn't increase its percent usage." Inhofe's remarks came during a hearing of the Subcommittee on Clean Air, Climate Change & Nuclear Safety. Inhofe noted that in 1997, when he became chairman of the subcommittee, the NRC "was more concerned about paperwork violations than real risk issues, and no one was even considering building new nuclear units." Now, NRC has moved to a risk-based regulatory climate and "people are actually talking about new nuclear generation," he said.

## *NUCLEAR NEWS FLASHES - Monday, May 23, 2005*

**--THE POSSIBLE NEED FOR A NEW U.K. NUCLEAR PRE-LICENSING REGIME** was touched on today by Prime Minister Tony Blair's chief scientific adviser, David King. Asked at a London energy conference for his thoughts on licensing should new nuclear construction be put back on the government's agenda, King noted the U.S. pre- licensing activity. He added, "I can see that there would be real advantages in going into a pre-licensing mode sooner rather than later in the U.K. We might even benefit from the current work being done in the U.S. on that." The U.K. dropped the idea of prelicensing reviews of Atomic Energy of Canada Ltd. and Westinghouse reactor designs in spring 2003 after the government's energy white paper backed energy efficiency and renewables to cut carbon dioxide emissions, relegating nuclear to an option that would be merely "kept open."

Handout #2, 1st handout, 6/1

Point Beach Nuclear Plant

# SUSTAINING IMPROVEMENT

Doug Cooper
Senior Vice President

NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant

## 95003 Inspection Findings

- Corrective Action Program
- Human Performance
- Emergency Preparedness
- Engineering/Operations Interface
- Engineering Design

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## Point Beach Nuclear Plant

### Site Excellence Plan

- Overhauled Comprehensive Excellence Plan
- Internal / External Assessments As Basis
- In Progress Prior To 95003 Inspection
- Resource Loaded
- Routinely Monitored and Updated

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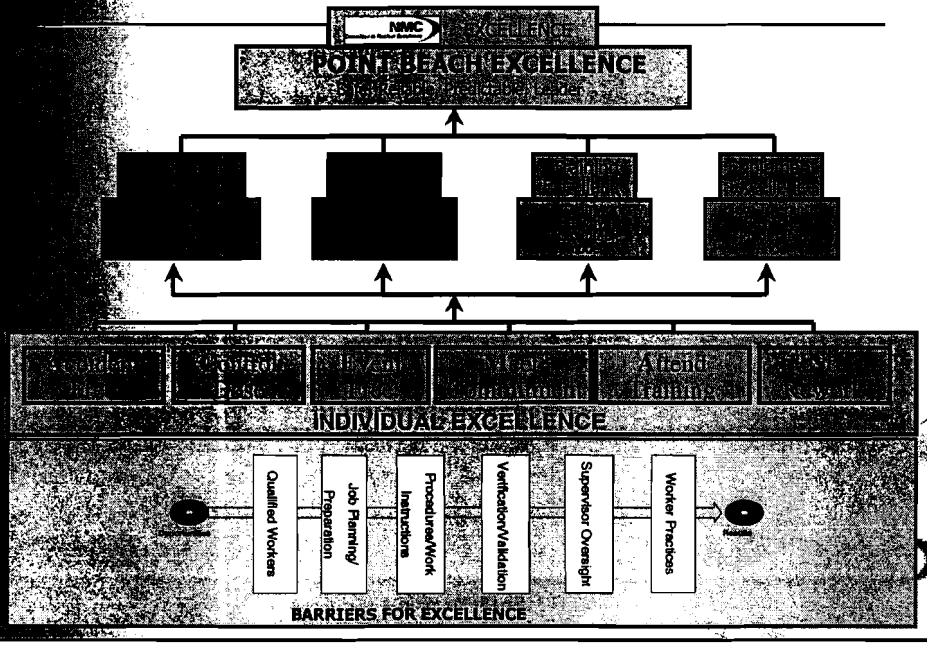
## Picture of Excellence



## Point Beach Nuclear Plant

### Six Critical Milestones
### "6 For Success"

- Dry Fuel Storage
- Spring Refueling Outage – 2005
- Confirmatory Action Letter (CAL) Closure
- Fall Refueling Outage – 2005
- Operations Training Program Accreditation
- INPO Evaluation

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## Point Beach Nuclear Plant

### Results To Date
### U2R27 Outage Goals

| | Goal / Actual |
|---|---|
| • Accident Free / Personnel Safety | |
| • Lost Time Accidents | 0 / 0 |
| • OSHA Recordable Injuries | 0 / 1 |
| • Radiological Events | <1 / 0 |
| • Event Free / Nuclear Safety | |
| • Unplanned Orange / Red Paths | 0 / 0 |
| • Reactor Trips (either unit) | 0 / 0 |
| • Safeguards Actuation (either unit) | 0 / 0 |
| • Loss of Shutdown Cooling | 0 / 0 |
| • Loss of Reactor Vessel Level Control | 0 / 0 |
| • Station Human Performance Clock Reset | 0 / 1 |

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## Point Beach Nuclear Plant

### Results To Date

- Successful Emergency Preparedness Exercise
- Human Performance-Related Issues Improved
- 134 OF 143 Confirmatory Action Letter (CAL) Items Completed
- 50 OF 65 Performance Measures In CAL Met Or Exceeded
- Engineering Related Calculations Longest Action To Complete
- Overall Improved Performance Observed in Excellence Plan
  - Oversight Assessments
  - Self-Assessments
  - INPO Reviews
  - Objective Performance Measures

NMC

## Point Beach Nuclear Plant

### NRC Public Comments On Point Beach

- Public Meetings– March 14 and April 19, 2005
  - Reiterated Importance Of CAL Completion and Outage Performance
  - Acknowledged Status of CAL Actions and Challenges in Calculation Project and EAL Revisions
  - Acknowledged Improvements In All CAL Areas
- Agency Action Review Meeting – May 25, 2005
  - Point Beach Continues To Operate Safely
  - Progress Being Made In All Areas Of Confirmatory Action Letter
  - Action Items Are Scheduled And Resource Loaded
  - Improvements Indicate Ability To Complete Commitments

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Handout #2, 2nd handout, 6/1

# Point Beach Nuclear Plant, Units 1 and 2 Discussion of Red Findings

Staff Presentation to the ACRS Plant
License Renewal Subcommittee
Patrick Louden, Branch Chief
Division of Reactor Projects, Region III
June 1, 2005

## Point Beach Red Findings

- Licensee identified Auxiliary Feedwater system finding in November 2001
- Two NRC Special Inspections performed:
  - December 2001 – February 2002
  - September 2002 – March 2003
- 2003 Inspection identified second Auxiliary Feedwater system finding

## Point Beach Red Findings

- Plant notified of final decision on first Red finding in April 2003
- Red Finding places Point Beach in Column IV of the NRC Action Matrix
- Second Red finding issued in December 2003

## NRC Inspection Procedure 95003

- Supplemental Inspection 95003 conducted from August to December 2003
  - Diagnostic in nature
  - Focused on known problem areas
  - Results determine any necessary additional NRC actions
- Point Beach 95003 inspection completed in three parts

## Point Beach 95003

- Teams comprised of inspectors from all NRC regional offices and included contractors
- Additional findings and violations identified
- Five general areas of concern identified through 95003 and baseline inspections

## NRC Areas of Concern

- Five areas of regulatory concern:
  - Human Performance
  - Engineering Design Control
  - Engineering/Operations Interface
  - Emergency Preparedness
  - Corrective Action Program

## Point Beach CAL

- These five areas formed basis for NRC Confirmatory Action Letter (CAL)
- CAL issued on April 21, 2004
- Licensee developed Commitment Letter based on their Site-Wide Excellence Plan
- Commitment Letter attached to CAL

## Point Beach Inspections

- Baseline team inspection membership was expanded
- Special inspections were conducted to evaluate the licensee's progress in implementing Confirmatory Action Letter items.

## Point Beach Current Performance

- Substantive Cross-cutting Issues
  - Substantive cross-cutting issues identified in the areas of Human Performance and the Problem Identification and Resolution
  - Also are identified as areas of concern in the Confirmatory Action Letter

## Point Beach Current Performance

- Human Performance has been a licensee focus and recent improvement has been noted
- Corrective Action Program as designed is sound
- Some aspects of implementation of the program need improvement

## Point Beach Current Assessment

- The licensee has made progress in all five Confirmatory Action Letter areas of concern
- The NRC focus is on sustainability of the licensee's corrective actions

Item # 3, handout 1 6/1

# Policy Issues Related to New Plant Licensing

Presented to
Advisory Committee on Reactor Safeguards

June 1, 2005

# Purpose of Presentation

- Brief Committee on the staff's recommendation on two policy issues
  - What shall be the minimum level of safety that new plants need to meet to achieve enhanced safety?
  - How shall the risk from multiple reactors at a single site be accounted for?
- Requesting approval on staff's recommendation on the two policy issues
  - Requesting approval in a letter

# Background/History

- SECY-03-0047, Policy Issues Related to Licensing Non-Light Water Reactor Designs, March 28, 2003
  - Staff recommended that implementation of enhanced safety through a process similar to that used in the evolutionary LWR and advanced light-water design certification reviews.
  - The staff also recommended that the following considerations be applied:
    - When using probabilistic or risk information, modular reactor designs should account for the integrated risk posed by multiple reactors necessary to achieve the overall electrical output desired.
    - The incremental risk to the surrounding population from adding additional units to an existing site is expected to be small due to the enhanced safety characteristics of new designs.

- SRM (to SECY-03-0047), June 26, 2003, direction on the policy issues:
  - Approved the staff's recommendation on implementation of the Commission's expectation for enhanced safety in future non-LWRs
    - Requested the staff to provide further details on the options for, and associated impacts of, requiring that modular reactor designs account for the integrated risk posed by multiple reactors

# Background/History (cont'd)

- In SECY-04-0157, "Status of Staff's Proposed Regulatory Structure for New Plant Licensing and Potentially New Policy Issues," the staff alerted the Commission to the new policy issue regarding level of safety
  - What level of safety should be the goal to achieve enhanced safety?
- SECY-05-0006, "Second Status Report of Staff's Proposed Regulatory Structure for New Plant Licensing and Policy Issues Related to Licensing Non-Light-Water Reactor Designs," dated January 7, 2005, the staff
  - The staff would submit final recommendations on integrated risk, containment performance, level of safety in mid 2005 to support pre-application reviews of new reactor designs
  - Resolution of level of safety and integrated risk are also fundamental to the development of the Framework

# General Guidelines in Assessing Options and Developing Recommendations

- Keep the risk to the population around a nuclear power plant site consistent with the Commission's 1986 policy statement on Safety Goals
- Utilize a risk-informed and performance-based approach, wherever practical, consistent with Commission guidance
- Use a technology-neutral approach
- Use the Commission's performance goals to assess the advantages and disadvantages of the options and to develop a recommendation
- Consider previous Commission guidance on these issues
- Consider the practicality of the options and recommendations

# Previous Commission Guidance (Policy Statements)

- Regulation of Advanced Nuclear Power Plants
  - "…. the Commission expects, as a minimum, at least the same degree of protection of the public and the protection of the public and the environment that is required for current-generation light water reactors. Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety …. The Commission also expects that advanced reactor designs will comply with this Commission's safety goal policy statement…."
- Severe Reactor Accident Regarding Future Designs and Existing Plants
  - "Although in the licensing of existing plants the Commission has determined that these plants pose no undue risk to public health and safety, this should not be viewed as implying a Commission policy that safety improvements in new plant design should not be actively sought. The Commission fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs."
- Safety Goals for the Operation of Nuclear Power Plants
  - "Current regulatory practices are believed to ensure that the basic statutory requirement, adequate protection of the public, is met."

# Two Policy Issues

- What shall be the minimum level of safety that new plants need to meet to achieve enhanced safety?

- How shall the risk from multiple reactors at a single site be accounted for?

# First Issue: Level of Safety Four Options Identified

- Option 1: Use current process
  - A minimum level of safety is not explicitly defined for achieving enhanced safety
  - A case-by-case determination of enhanced safety is made
  - A minimum level of safety would not be specified in the technology-neutral framework for new plant licensing

# First Issue: Level of Safety Four Options Identified (cont'd)

- Option 2: Define the minimum level of safety as the Quantitative Health Objectives (QHOs)
  - The QHOs (as expressed in the Commission's Safety Goal Policy) are selected as the minimum level of safety to demonstrate that enhanced safety has been achieved for new reactor designs
  - The QHOs would be used to assess whether new reactor designs, currently under review, meet the Commission's expectation of enhanced safety
  - The framework would be developed based on a minimum level of safety defined by the QHOs
    - The technology-neutral regulations would be developed to achieve at least the safety goal level of safety

# First Issue: Level of Safety Four Options Identified (cont'd)

- Option 3: Develop other risk objectives for the minimum level of safety
  - Other risk objectives would be developed for the QHOs on a technology-neutral basis similar to the subsidiary risk objectives, core damage frequency (CDF) and large early release frequency (LERF), developed for the current operating LWRs
  - Other risk objectives would be used to assess whether new reactor designs, currently under review, meet the Commission's expectation of enhanced safety
  - The framework would be developed based on a minimum level of safety defined by these new risk objectives

# First Issue: Level of Safety Four Options Identified (cont'd)

- Option 4: Develop new Quantitative Health Objectives for the minimum level of safety
  - New QHOs, which are more stringent than the ones defined the Commission's Safety Goal policy statement, would be developed
  - The QHOs would be used to assess whether new reactor designs, currently under review, meet the Commission's expectation of enhanced safety
  - The framework would be developed based on a minimum level of safety defined by the QHOs

# Option 1: Advantages & Disadvantages

## *Use Current Process*

Advantages

- No changes needed to current regulatory practice
- Provides maximum flexibility

Disadvantages

- Not necessarily result in technology-neutral, risk-informed or performance-based
- Not clear that supports Commission's expectations
- Could lead to different results
- May be more likely to be challenged by stakeholders
- Not readily scrutable
- Relies on subjective judgment
- Likely not to result in consistency and uniformity
- Likely not to promote stability and predictability

# Option 2: Advantages & Disadvantages

## *Define Safety Level as the QHOs*

Advantages

- Eliminates need to develop criteria on a case-by-case basis
- Provides for a more efficient, scrutable and objective regulatory process
- Increases the level of consistency and uniformity
- Promotes regulatory stability and predictability
- Defines safety level that is technology-neutral, risk-informed and performance-based
- Implements Commission's expectations
- Allows flexibility, allows applicant to propose other risk measures

Disadvantages

- In providing the flexibility, may result in inconsistent approaches for similar designs in an increase in resources
- Level 3 PRA needed

# Option 3: Advantages & Disadvantages

## *Define other Risk Measures as the Safety Level*

Advantages

- Eliminates need to develop criteria on a case-by-case basis
- Provides for a more efficient, scrutable and objective regulatory process
- Ensures consistency and uniformity
- Promotes regulatory stability and predictability
- Defines safety level that is technology-neutral, risk-informed and performance-based
- Implements Commission's expectations
- Level 3 PRA not needed

Disadvantages

- Significant uncertainties regarding feasibility
- Not apparent that technology-neutral risk objectives tied to the QHOs can be developed
- LWR surrogates: CDF and LERF
  - Definition of core damage, for example, does not apply to the PBMR
- CDF and LERF quantitative objectives (1E-4/ry and 1E-5/ry, respectively)
  - Based on experience gained from substantial number of Level 3 LWR PRAs
- Data and experience needed
- Large increase in resources

# Option 4: Advantages & Disadvantages

## *Develop More Stringent QHOs as the Safety Level*

Advantages

- Eliminates need to develop criteria on a case-by-case basis
- Provides for a more efficient, scrutable and objective regulatory process
- Ensures consistency and uniformity
- Promotes regulatory stability and predictability
- Defines safety level that is technology-neutral, risk-informed and performance-based
- Implements Commission's expectations

Disadvantages

- Require considerable time and staff resources
- Need to identify type of analysis (or even develop new method) to demonstrate that these new QHOs have been met
- Likely to have significant concerns from external stakeholders
- Significant uncertainties regarding feasibility

# Recommendation for Minimum Level of Safety

- Recommend that Option 2 be selected
  - ***The QHOs should be the minimum level of safety for new plant licensing in achieving the Commission's expectation for enhanced safety***
- Provides for a uniform application of safety level across all reactor technologies
- Promotes regulatory stability and predictability for new plant licensing
- Defines safety level that is technology-neutral, risk-informed, performance-based
- Implements Commission's expectations
- Consistent with level of safety adopted by the industry
- Allows greater flexibility to both staff and the applicant
- Feasible with the least impact on time and staff resources

# Second Issue: Integrated Risk Three Options Identified

- Option 1: No consideration of integrated risk
  - The risk information is evaluated on a per reactor basis, not a per site basis
  - Integrated risk from new reactors would not be considered
  - Technology-neutral framework would not consider integrated risk

# Second Issue: Integrated Risk Three Options Identified (cont'd)

- Option 2: Quantification of integrated risk at the site solely from new reactors
  - The integrated risk solely associated with new reactors at a site would not exceed the risk expressed by the QHOs
  - For ongoing design reviews, staff would ensure that their integrated risk not exceed the risk expressed by the QHOs
  - Criteria and guidelines in technology-neutral framework developed so that the integrated risk from new reactors not exceed the risk expressed by the QHOs

# Second Issue: Integrated Risk Three Options Identified (cont'd)

- Option 3: Quantification of integrated site risk
  - The integrated risk from both existing and new reactors would not exceed the risk expressed by the QHOs
  - For ongoing design reviews, staff would ensure that the integrated risk from both the existing and new reactors not exceed the risk expressed by the QHOs
  - Criteria and guidelines in technology-neutral framework developed so that the integrated risk from both existing and new reactors not exceed the risk expressed by the QHOs

# Option 1 Advantages & Disadvantages

***No Quantification of Integrated Risk***

Advantages

- Status quo is maintained with no consideration of integrated from multiple reactors at same site
- Least impact on time and staff resources

Disadvantages

- No quantitative assessment of or limit on integrated risk
- Risk to population near a site could be greater than the QHOs
- May be difficult to justify and communicate to the public

# Option 2 Advantages & Disadvantages

***Quantification of Integrated Risk of New Reactors Only***

Advantages

- Qualitatively assesses and limits the total integrated site risk
- The integrated risk from new reactors would not exceed the risk expressed by the QHOs
- Preserves the current regulatory stability and predictability
- Consistent with Commission's statements

Disadvantages

- Full quantitative assessment of the integrated site risk is not evaluated
- Increases COL action items arising from ESPs and design certifications
- Public may find it difficult to understand why existing reactors should not be held to the higher minimum level of safety

# Option 3 Advantages & Disadvantages

***Quantification of Integrated Risk of both Existing and New Reactors***

Advantages

- Quantitatively assesses and limits the total integrated site risk
- The integrated risk from both existing and new reactors would not exceed the risk expressed by the QHOs
- Relatively straightforward to communicate to the public

Disadvantages

- Disrupts the current regulatory process
- Significantly increases COL action items arising from ESPs and design certifications
- Goes beyond Commission's expectations
- Generic site may not bound the actual site
- May not be workable for construction of new reactors at existing sites if applicant unrelated to current licensee

# Recommendation on Integrated Risk

- Recommend that Option 2 be selected
  - ***The integrated risk solely associated with new reactors at a site not exceed the risk expressed by the QHOs***
- Existing reactors, under current regulatory structure, found to pose no undue risk to public health and safety
- Ensures that any incremental risk to the site is insignificant
- Defines integrated risk criteria that is technology-neutral, risk-informed, performance-based
- Implements Commission's expectations
- Promotes regulatory stability and predictability for new plant licensing while preserving the current regulatory structure

# In Summation

- Requesting letter from Committee
  - approving staff recommendation on the two policy issues, level of safety and integrated risk
- Staff SECY paper due to the Commission June 30, 2005

Item # 4, handout 1, 6/1

# EPRI/NRC-RES FIRE PRA METHODOLOGY

J.S. Hyslop, NRC/RES
Steven P. Nowlen, SNL
Bijan Najafi SAIC (for R. Kassawara, EPRI)

ACRS Full Committee
June 1, 2005
Rockville, MD

*A Collaboration of U.S. NRC Office of Nuclear Regulatory Research (RES) & Electric Power Research Institute (EPRI)*

## BACKGROUND

- MOU between NRC-RES and EPRI on fire risk
- One of several elements on MOU
- Primary objective of this program: develop, field test, and document state-of-art
- Prior briefings of ACRS, including focused briefing in May 05
- Purpose: Brief ACRS on final NUREG/CR-6850, EPRI 1008239 "*EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*" which addresses public comments

## ROLES OF PARTICIPANTS

- NRC-RES and EPRI develop and test methods
- Three volunteer pilot plants support testing
- Other participating licensees provide peer-review of methods
  - Peer review indicated that step change produced in many areas of fire PRA methodology
- EPRI and NRC-RES reach consensus on documented methodology



## EXPECTED USE OF METHODOLOGY

- Support for new rule 10CFR50.48c implementation
- Analyses under the current fire protection regulations (i.e. exemptions/deviations or other plant changes such as risk-informed technical specifications)
- Basis for staff review guidance that RES will develop for NFPA 805 related changes
- ANS fire risk standard
- Analysis and reviews of fire protection inspection findings (phase 3 SDP)

## ADVANCEMENT TO STATE OF ART

- Improvements made in areas important to fire risk (resource constraints considered)
- Means to advance
  - Consolidate existing research
  - Analyze more extensive data
  - Modify existing methods
  - Develop new approaches



## RELATIONSHIP TO FIRE MODEL V&V

- Fire modeling tools provide input to fire PRA
- Fire model verification and validation (V&V) is required for NFPA 805 applications
- In limited cases, fire models (empirical correlations) utilized
  - Address cases where computational fire models inadequate
  - Fill important gaps in fire PRA
- PRA Methodology document not a reference for fire models
  - Any necessary V&V left to analyst

## PUBLIC COMMENTS

- Comments provided during public comment period by industry and consultants
  - Duke Power, Florida Power and Light, EPM, RDS
- Comments provided by NRR
- No public comment required NRC-RES and EPRI to significantly adjust our approach
  - Few comments on state-of-the-art limitation
  - Remaining comments were minor and clarifications



## MILESTONES

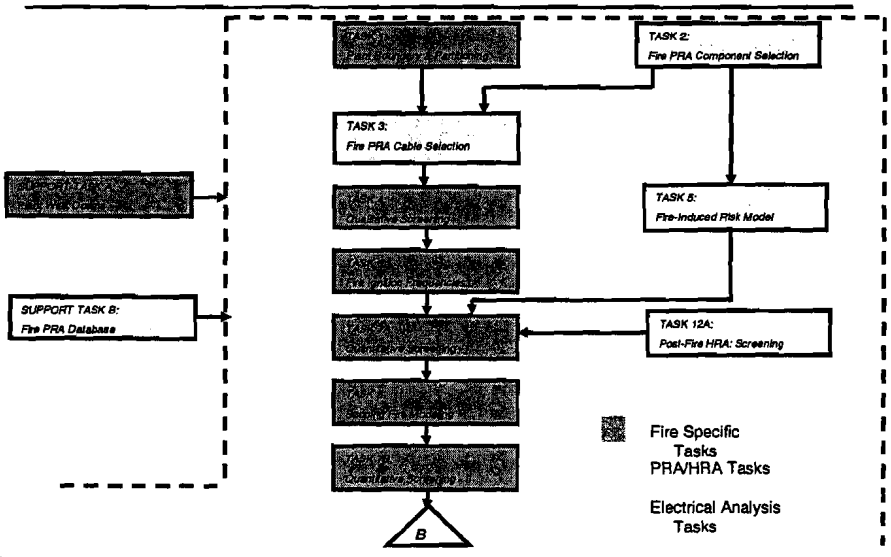
| | |
|---|---|
| • Draft report for public comment | Oct 2004 |
| • ACRS | May/Jun 2005 |
| • Public Fire PRA Methodology Workshop | Jun 14-16, 2005 |
| • Publication | Aug 2005 |
| • BWR pilot | 2006 |
| • Revision of methodology (if needed) | Dec 2006 |

# PROJECT TEAM

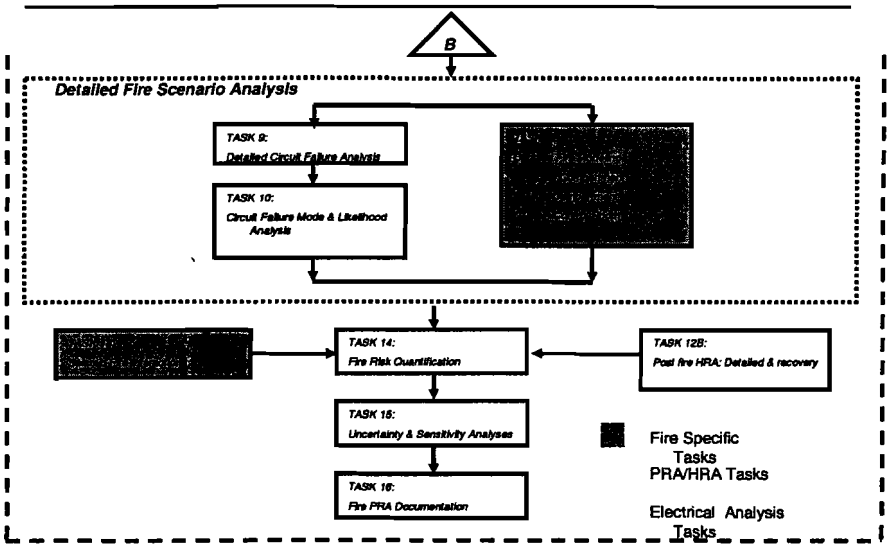
- Covers all technical disciplines critical to Fire PRA
  - Technical Lead: B. Najafi, S. Nowlen
  - General PRA & plant systems analysis: A. Kolaczkowski, R. Anoba
  - Circuit Analysis and Appendix R: D. Funk, F. Wyant
  - Human Reliability Analysis: J. Forrester, W. Hannaman, A. Kolaczkowski
  - Fire analysis: F. Joglar, M. Kazarians
  - Consultants: A. Mosleh, D. Bley
- Collectively, over 250 years of relevant experience
- Principal authors of documented Fire PRA methods in the US for the past 2 decades
- Experience with use of previous methods; their strengths and weaknesses
- The Methodology reflects the consensus of this team, EPRI and RES



# FIRE PRA PROCESS FLOW CHART



# FIRE PRA PROCESS FLOW CHART






# Task 2: Fire PRA Component Selection

- Determines what will be credited in Fire PRA model
- Advances over IPEEE model
  - Multiple spurious actuation events
  - Key instrumentation supporting post fire operator actions
- Public comments
  - Added search for "new" scenarios and associated components
  - Added more on unique manual actions including supporting instrumentation needed as well as accounting for equipment effects as a result of actions
- Cable selection (**task 3**) performed for all fire PRA components
- Fire induced risk model (**task 5**) models these issues

## Task 12: Post Fire HRA

- Main advances
  - Screening level Human Error Probabilities (HEPs)
    - Range from 10x Internal Events PRA HEPs to 1.0
  - Identification and discussion of performance shaping factors (PSFs) for detailed analysis
    - Plant and scenario-specific PSFs for estimating best estimate HEPs for significant fire scenarios
- Procedure does not provide detailed quantification guidance
- Public comments
  - Removed discussion of fire-specific pre-initiator human failure events (HFEs)
    - Those impacting fire protection systems, barriers, general fire protection program elements
    - Possible confusion/overlap with use of experience/data covered in other Tasks
    - Does not preclude plant-specific HRA of fire-specific pre-initiator HFEs
  - Added "general" guidance on use of existing HRA methods, BUT no specific quantification guidance as requested by one comment



## Task 6: Fire Frequencies

- *Fire Frequencies*: significant improvements made here
  - Most fire sources now use component-based frequency
    - Allows for more consistent, refined, and reasoned compartment and scenario frequencies that reflect plant specific configuration
  - Extensive analysis of event data
    - IPEEEs typically used full unscreened event set for frequency and applied generic severity factors
    - We screened events for risk-relevance (potentially challenging)
    - We also utilize fire severity profiles that have implicit links to the final frequency event sets (more on profiles later)
  - This area was the subject of much discussion and adjustment during peer review
  - Several public comments requested clarification of specifics, no major changes

## Task 9: Detailed Circuit Analysis

- Generally reserved for cases in which quantitative screening indicates a clear need and advantage for further analysis
- Detailed failure modes analysis
- Objective is to screen out cables that cannot impact the ability of a component to complete its credited function
- Public comments
  - Enhanced "risk-informed" guidance to focus analysis on failure modes of concern
  - Incorporated guidance for human factors interface



## Task 10 – Circuit Failure Mode Likelihood Analysis

- Probabilistic Based Circuit Analysis
- Two Methods Presented
  - Expert Panel Results
  - Computation-Based Analysis
- Requires Knowledge About Circuit Design, Cable Type and Construction, Installed Configuration, and Component Attributes
- Generally Reserved for Only Those Cases that Cannot be Resolved Through Other Means

## Task 10 – Circuit Failure Mode Likelihood Analysis

- Key Insights
  - Our Knowledge is Greatly Improved but Uncertainties are Still High
  - Practical Implementation is Challenging
  - Further Analysis of Existing Test Data and Follow-On Tests Would be Beneficial
- Public and Peer Review Comments
  - Extensive Discussions Regarding the Most Appropriate Way to Tally Spurious Actuation Probabilities (Many Subtleties for Implementation)
  - Team's Consensus is that Expert Panel Values are generally Conservative as currently applied
  - Additional Independent Review of the Circuit Analysis Method was Solicited (Favorable, but limitations acknowledged)

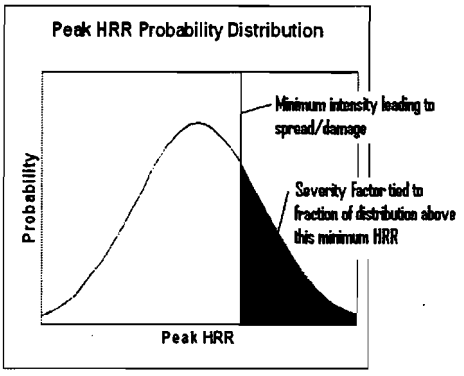


## Task 11: Detailed Fire Modeling

- Scope: Define and evaluate specific fire scenarios
  - Single, multi-compartment, main control room fire scenarios
- General approach follows traditional pattern:
  - Identify fire sources and targets
  - Fire growth/spread/damage analysis including fire severity
  - Fire detection/suppression analysis
  - Final output is conditional probability of fire consequences given fire ignition
- Special models developed to account for NPP fire scenarios beyond the capability of existing computational fire models

## Task 11: Detailed Fire Modeling Heat Release Rate and Severity

- Previous methods (e.g. IPEEE) used a fixed HRR and severity factor
- New approach ties severity factor to a distribution on peak fire intensity
- HRR distributions developed for various ignition sources
  - Expert judgment based upon evidence from events and data
  - Captures high intensity fires
- Basis for scoping fire modeling (**Task 8**) which eliminates components from fire model






## Task 11: Detailed Fire Modeling Special Models

- High energy arcing faults (new)
  - Critical to switchgear room fire risk
  - An empirical rule set type model based on operating experience
  - High-energy phase, defined by a "zone of influence"
  - Thermal phase (enduring fire) treated like other fire sources
- Main control board (new)
  - Critical to control room fire risk
  - A probabilistic model for fire propagation inside the main control board
- Cable fires (modified from IPEEE approaches)
  - Critical to cable spreading room and cable tunnel fire risk
  - Fire spread in a single tray or cable tray stacks

## Task 11: Detailed Fire Modeling Other Special Models

- Fire propagation to adjacent cabinets (consolidation)
- Passive fire protection features (consolidation)
- Hydrogen fires (new)
- Turbine generator fires (new)
- Smoke damage (consolidation)



## Task 11: Detailed Fire Modeling Detection & Suppression Analysis

- Probability of Non-Suppression = conditional probability that fire lasts long enough to cause postulated damage
- Approach credits:
  - Prompt detection & suppression (by plant personnel or fire watch)
  - Automatic detection and suppression
    - Reliability and availability
    - Effectiveness (scenario-specific)
  - Manual detection
  - Manual suppression by fire brigade
    - Model based on operating experience – fire suppression time curves
- Improvements over previous methods:
  - More rigorous review/analysis of event data – long lasting fires
  - Explicit calculation framework (event tree)

## Public Comments on Task 11

- Editorial/clarification comments including consistency with fire protection SDP, and NEI 04-02
- V&V of fire models
  - NFPA 805 requires that fire models are verified and validated.
  - Our report documents fire PRA state-of-the-art – broader applicability
  - "Models" are cited when team consensus concluded need is critical, and identified method represents a reasonable approach and/or current best practice
    - e.g., the "special models" discussed previously
  - We did not V&V recommended approaches



## Task 15: Uncertainty and Sensitivity Analyses

- Addresses the *process* for uncertainty and sensitivity analyses
  - Modeling and data uncertainties
  - A comprehensive list of *specific* uncertainty sources for each task has been developed
  - Explicit guidance on quantification (e.g., uncertainty bounds) for each identified source is NOT provided
    - May be able to add as more demonstrations are performed
- Some changes were made as a result of public comments
  - Consolidated discussions of uncertainties from individual tasks under Task 15 (clarification issue)
  - Separated uncertainties to be addressed from technical quality issues
    - Discussion of both remains – see Appendix V
  - Added discussion on usefulness of sensitivities for screened compartments
  - Other minor clarifications and editorial comments included

## INSIGHTS

- CDF Insights (in the author's judgment compared to IPEEE)
  - Expect overall range of CDF for fleet of plants to be maintained
  - Expect individual risk profile of some plants to change
    - Multiple spurious actuations, high energy arcing faults
    - Control room
  - Continued use of this methodology needed
- Cable tracing to support fire PRA still major resource requirement
  - Address via iterative, screening nature of fire PRA



## STATE OF ART IN FIRE PRA

- Best available method to estimate fire risk and obtain insights.
- Improvements will benefit state-of-the-art
  - Spurious actuations
  - Post-fire HRA
  - Low power and shutdown operations
  - Plant-specific assessment of fire fighting

Item #5, handout 1, 6/1

# STATUS OF EVALUATION OF A BROADER CHANGE TO SINGLE FAILURE CRITERION

## PRESENTATION TO ACRS


**Hossein Hamzehee (415-6228)**
**John Lane (415-6442)**
**Office of Nuclear Regulatory Research**


**June 1, 2005**

## Status of Evaluation of Broader Change to Single Failure Criterion

**Purpose:**

- **The purpose of this presentation is to provide a summary of:**
  - **Status of our evaluation of a broader change to Single Failure Criterion.**
  - **Planned follow-up activities.**
- **We are seeking ACRS to provide feedback (via a letter) on:**
  - **Work completed to date on risk-informed alternatives to current Single Failure Criterion (SFC)**
  - **Planned follow-up activities**

# Evaluation of Broader Change to Single Failure Criterion

## Presentation outline:

- Background
- Summary of Technical Approach and Work Completed To date
- Summary of NRR Major Comments
- Planned Follow-up Activities
- Schedule

## Broader Change to Single Failure Criterion

**Background:**

- In March 31, 2003 SRM on risk-informed changes to 10 CFR 50.46 (ECCS Acceptance Criteria), the Commission approved most of staff recommendations on possible changes to LOCA requirements, and also directed the staff to:

  - Risk-inform current requirements of coincidental LOCA/LOOP.
  - Pursue a broader change to SFC and inform the Commission of its findings beyond what is being considered for exemption request for LOCA/LOOP.

- This work was done in response to Commission directive to "Pursue a Broader change to SFC"

  - Our interpretation of "Broader Change" is RI alternatives that could apply to all plant functions (safety and non-safety).
  - Could lead to changes in licensing, programmatic activities (e.g., testing and inspections), and plant performance monitoring.

## Broader Change to Single Failure Criterion

### Background (Cont'd):

- **The term "Single Failure" is defined in 10 CFR Part 50 Appendix A:**

  **"A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single-failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions."**

  **[Associated Footnote: "Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development."]**

## Broader Change to Single Failure Criterion

**Background (Cont'd):**

- SFC requirements exist in two major contexts:
  - GDC of 10 CFR 50 Appendix A which identify safety functions and associated safety systems to which SFC applies.
  - Design-Basis Accident (DBA) guidance of chapter 15 of RG 1.70 and Standard Review Plan (SRP)
- SFC is one element of NRC defense-in depth philosophy
  - Accomplishment of key safety functions should not be dependent on a single element of design, construction or operation.
  - SFC promotes high safety system (safety function) reliability
  - Other regulations, guidelines, and programs, along with SFC, promote high reliability; e.g., QA requirements, Tech. Specifications, Testing, Maintenance requirements, Inspections

## Background (Cont'd):

- Application of SFC has sometimes led to redundant system components having low risk significance based on PRAs.
  - DEG LOCA coincident with LOOP and a DG failure.
  - Application of worst single failure assumption for DBAs may result in unnecessary constraints on licensees.
- SFC has not always been applied uniformly to passive components in fluid systems
  - A risk-informed alternative should also consider potential failure of passive components
- Application of SFC has not always led to safety system design with reliabilities that were commensurate with their safety significance.
  - SFC supplemented by other NRC guidance, rulemaking and programs contributed significantly to maintaining acceptable level of safety; e.g., SBO rule, ATWS rule, etc.

# Policy Issues

- A proposed risk-informed and performance-based alternative to SFC need to be consistent with Commission PRA regulatory policy guidance.

  - Maintaining defense-in-depth
  - Maintaining adequate safety margin
  - Security constraints
  - Consideration of uncertainty

- A proposed risk-informed and performance-based alternative to SFC need to be consistent with Commission guidance on phased approach to PRA quality.

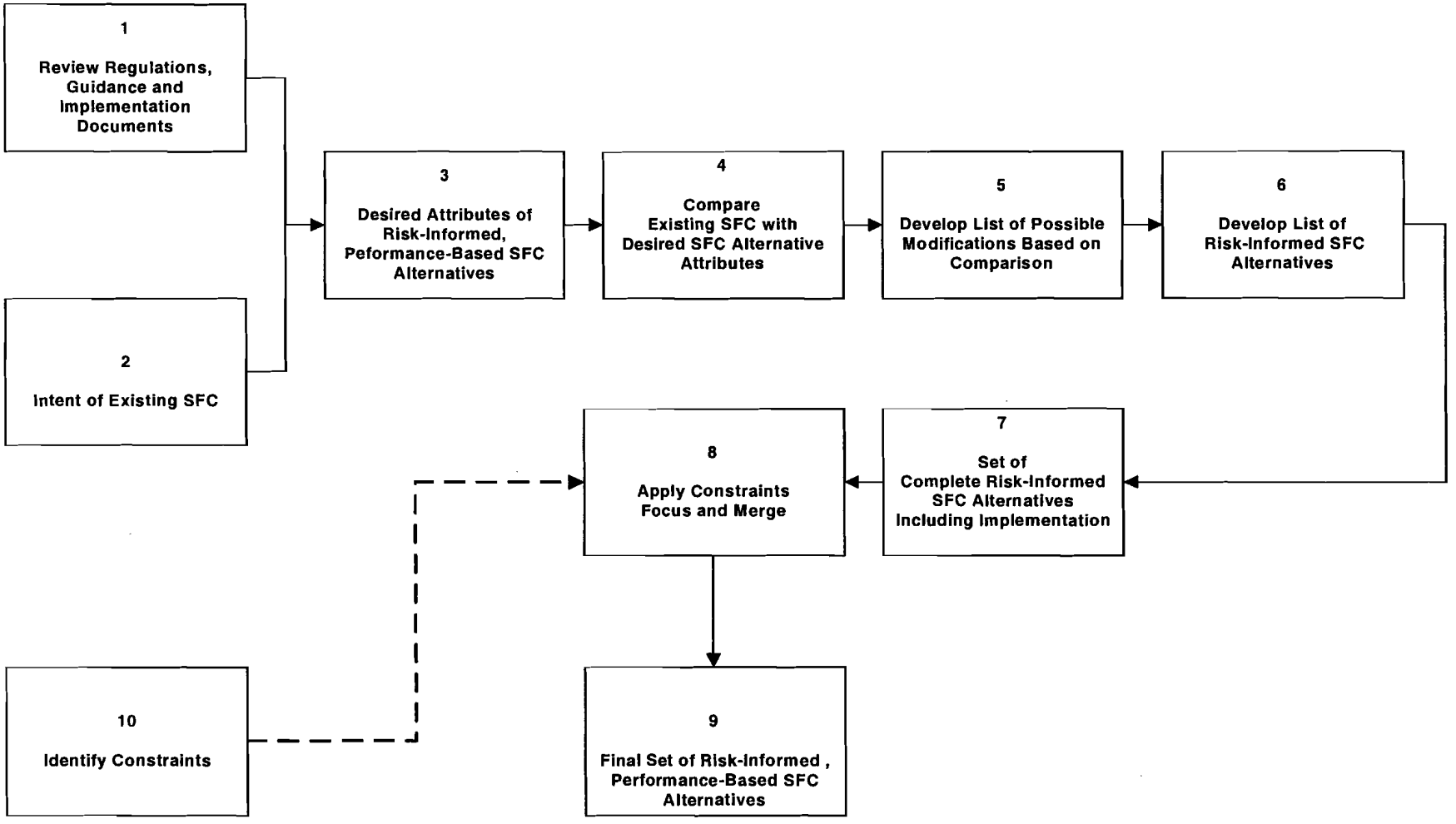
- A proposed risk-informed and performance-based alternative to SFC need to be consistent with Commission backfit and regulatory analysis policy.

- A proposed risk-informed and performance-based alternative to SFC need to be consistent with other on-going risk-informed activities; examples are:
  - Risk-informing requirements of 10 CFR 50.46
  - LOCA/LOOP exemption request from BWROG
  - Technical Specifications improvement

## Summary of Technical Approach and Evaluation Results

**Evaluation Process:**

- A process was developed to identify and evaluate potential risk-informed and performance-based alternatives to SFC
  - This process is shown in a flowchart
- One major step of the process was to define desired attributes that a potential risk-informed and performance-based alternative should have. These attributes are:
  - Provide functional reliability
  - Maintain defense-in-depth
  - Risk-inform application of SFC in design-basis safety analysis
  - Use performance-based regulatory approach
  - Amenable to effective implementation
  - Coherent with other risk-informed regulatory initiatives
  - Maintain design requirements that contribute significantly to plant built-in capability to resist security threats




# SFC Alternative Process Flowchart

## Summary of Technical Approach and Evaluation Results

**Baseline Alternative:**

- This alternative continues to make risk-informed changes to regulatory requirements that involve specific issues, such as:
  - Proposed rulemaking on 10 CFR 50.46
  - LOCA/LOOP requirements
  - Risk-informed Tech. Specifications initiatives

- This alternative could result in measurable progress over time (one issue at a time) rather than attempting to address many related issues and regulations simultaneously.

- This alternative would not include a broad change to current licensing requirements to risk-inform SFC.

- Improvement of coherence among risk-informed programs would be limited under this alternative.

- This alternative would also consider updating the footnote in 10CFR50 Appendix A on passive failures in fluid systems.

## Summary of Technical Approach and Evaluation Results

**Alternative 1:**

- **This alternative would risk-inform failure assumptions made in DBA analyses in Chapter 15 of FSAR.**

  - **Single failures resulting in sequences with sufficiently low frequency would no longer be required in DBA analysis.**
  - **Unlikely initiating events would also be eliminated from DBA analysis.**
  - **Multiple failures in sequences with sufficiently high frequency would be added to DBA analysis.**
  - **Plant PRA will be used to demonstrate that cumulative frequency of sequences excluded from DBA is small.**

- **Potential plant changes proposed under Alternative 1 must also satisfy RG 1.174 guidelines.**

  - **This alternative could result in additional predicted safety margin.**
  - **Examples of potential changes are core peaking factors, power uprate, EQ requirements, allowed outage times**

## Summary of Technical Approach and Evaluation Results

**Alternative 1 (Cont'd):**

- Quantitative frequency criteria would be established for removal/addition of event sequences in DBA analysis.

- Proposed rule for 10 CFR50.46 could be considered special case of Alternative 1

  - SFC not required for LOCA sizes greater than Transition Break Size

- This alternative would be consistent with other ongoing risk-informed activities

  - 10 CFR 50.46
  - LOCA/LOOP
  - Framework for advanced reactors

**Alternative 2:**

- This alternative would risk-inform application of SFC to safety-related systems based on their safety significance.

- Risk-informed process proposed here takes advantage of current safety categorization process used in 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components."

  - RISC-1 Systems: Safety Related & Safety Significant
  - RISC-2 Systems: Non-Safety Related & Safety Significant
  - RISC-3 Systems: Safety Related & Low Safety Significant
  - RISC-4 Systems: Non-Safety Related & Low Safety Significant

- This alternative includes 3 sub-alternatives based on level of defense-in-depth desired for RISC-3 systems.
  - Alternative 2a: Redundant safety-related train(s) would be removed from service
  - Alternative 2b: Redundant safety-related train(s) would be re-classified as non-safety-related
  - Alternative 2c: Operational flexibility would be provided for redundant train(S)

## Summary of Technical Approach and Evaluation Results

**Alternative 2 (Cont'd):**

- Risk-informed requirements would be established for each risk category .
- Performance monitoring of system reliability would be required.
- Potential plant changes proposed under Alternative 2 must also satisfy RG 1.174 guidelines.
- This alternative promotes coherence with other risk-informed regulatory initiatives; e.g., 10CFR50.69, 10CFR50.46, etc.
- Implementation of this alternative is expected to require significant effort by NRC and industry.

## Summary of Technical Approach and Evaluation Results

**Alternative 3:**

- This alternative would generalize SFC by applying a combination of:

  - Quantitative top-level risk targets (CDF and LERF)
  - Lower-level functional reliability targets commensurate with challenge frequency (e.g., post-trip decay heat removal function)
  - Requirements (targets) on redundancy and diversity

- This alternative would allow Licensees to determine which plant features to credit to meet targets.

  - Regulatory attention would be on plant-specific performance weakness relative to established targets.

- Extra licensee and regulatory attention would be directed to areas where plant redundancy and diversity do not meet targets.

## Summary of Technical Approach and Evaluation Results

**Alternative 3 (Cont'd):**

- **Performance monitoring would be required to determine whether performance targets are met, per regulatory guidance such as NUREG/BR-0303, "Guidance for Performance-Based regulation."**
  - **This applies to all 3 alternatives.**
- **This alternative promotes coherence with other risk-informed regulatory initiatives, such as Reactor Oversight Process.**
- **Implementation of this alternative is expected to require significant effort by NRC and industry.**

## Summary of Technical Approach and Evaluation Results

**Conclusions:**

- **This work has identified and evaluated a range of risk-informed alternatives to current Single Failure Criterion.**

- **Additional evaluation and stakeholder involvement are necessary to assess practicality of implementing these alternatives**

  - **Interactions with stakeholders could result in identification of other viable alternatives.**

- **At this time, the staff does not recommend one alternative over another until follow-up activities are completed.**

## Broader Change to Single Failure Criterion

**Planned Follow-up Activities:**

- **In response to a May 9, 2005 SRM, the staff will develop a formal program plan to make risk-informed and performance-based revision to 10 CFR Part 50.**
  - **The staff plans to continue follow-up activities and will include them in this formal program plan**
  - **This approach ensures that any changes to SFC are evaluated in a broader context with other risk-informed changes to 10 CFR Part 50.**
- **Planned follow-up activities include additional evaluation of implementation issues and additional stakeholder involvement.**

## Summary of NRR General Comments

- NRR has provided general and specific comments on the draft report that must be considered before the report becomes final. RES is working to resolve these comments as practicable. General comments from NRR are summarized below.

- It may be more appropriate to postpone further effort and include it in the "formal program plan" discussed in May 9, 2005 SRM to make risk-inform and performance-based revisions to 10 CFR Part 50.

- Overly broad replacement of SFC for currently licensed plants is not prudent. Efforts to risk-inform SFC need to proceed cautiously and systematically, with clear understanding of potential safety and resource impact. Therefore, the report should focus more on pros and cons of broadening the relaxation of SFC versus not doing so.

- When the report is forwarded to the Commission, it should be made clear that it may be too early to recommend a specific alternative, and that outstanding technical issues exist which need to be resolved.

## Summary of NRR General Comments (Cont'd)

- **The report states that SFC is proxy for reliability. Other benefits of SFC include avoiding excessive reliance on a particular element of plant safety, maintaining DBA mitigating capability during maintenance. These other benefits need to be addressed more directly in the alternatives.**

## Broader Change to Single Failure Criterion

**Schedule:**

- Draft technical report was completed in February 2005
- Draft technical report was issued for inter-office review/comment in March 2005 and comments were received in May 2005.
- ACRS briefing in June 2005
- Receive ACRS feedback via a letter by June 30, 2005
- Issue a Commission paper (with technical report) by July 29, 2005

# Evaluation of Broader Change to the Single Failure Criterion

SFC Alternative Examples

# Alternative 1
# Example Description

- Illustrates potential for removal from design-basis an operationally-limiting, low frequency DBA sequence that includes an initiating event with a single failure

# Alternative 1 Example
# Key Features & Application

- Simplified containment event tree constructed for a double-ended MSLB in a PWR
- Limiting containment pressure event sequence: MSLB at low power with SFC = loss of DC bus
- Estimates for the IE frequency, low power probability and loss of DC probability, were obtained from typical PWR data

# Alternative 1 Example Results

- Limiting DBA event sequence has frequency of 5E-10 /yr. Containment pressure for this event sequence is 41.85 psig (design pressure 42 psig)

- If all single failure sequences were removed from the MLSB analysis on the basis of low frequency, peak predicted containment pressure would be in the range 32-36 psig, therefore larger margins predicted

- Illustrates instance where low frequency event sequence, including SFC, may limit plant operations. Alternative 1 could provide more operational or performance flexibility

# Alternative 2 Example Description

- Illustrates:
  - Potential for relaxation of SFC requirements for non-safety-significant, safety-related systems (RISC 3)
  - Enhancement of safety by now requiring monitoring of safety-significant, non-safety-related systems (RISC 2)

# Alternative 2 Example
# Key Features & Application

- BWR/4 PRA SPAR model plus 50.69 safety classifications used to classify all plant systems

- Use RAW importance measure to identify safety-related/low safety-significant systems and non-safety-related/safety significant systems

- 3 risk-informed sub-alternatives for relaxation of the SFC requirements are applied, including potential removal of redundancy requirement for low safety-significant systems. The impact on ΔCDF was evaluated.

# Alternative 2 Example Results

- Four systems categorized as safety-related and low-safety-significant (RISC-3): Emergency Cooling Water, Standby Liquid Control, Core Spray, Reactor Building and Closed Cooling Water
- Bounding evaluation reduced all 4 RISC-3 systems to single train; $\Delta CDF = 1.2E\text{-}5$ /yr
- 3 RISC-3 systems reduced to single train (SLC, CS, RBCCW) $\Delta CDF = 4.1E\text{-}7$ /yr
- 2 non-safety related, safety-significant (RISC-2) systems (PCS, TBCCW) would require monitoring of reliability
- Illustrates potential for relaxing SFC requirements with small impact on plant risk and for increased safety as result of RISC-2 monitoring

# Alternative 3 Example Description

- Illustrates
  - use of graded redundancy requirements based upon initiating event category
  - Guidance on diversity, and
  - Supplementation of SFC with top-level risk guidelines and safety function reliability targets

# Alternative 3
# Key Features & Application

- PWR PRA SPAR model
- Loss of offsite power initiating event (.03/yr)
- Decay heat removal safety function
- Based upon assignment of IE category "frequent" (>1E-2/yr)
  - A double-failure requirement specified for DHR
  - DHR functional unreliability target specified 1E-4
  - Overall plant CDF target specified 1E-4/yr
- Demonstrates how different combinations of DHR systems may be credited to satisfy the targets

# Alternative 3 Example Results

- CDF (total, not just LOOP contribution) and DHR function unreliability computed for several combinations of DHR systems
- {EFW only} satisfies SFC, but none of the new targets
- {EFW + any other train} satisfies all targets
- {EFW + all other trains} provides increased flexibility
- Illustrates how plant system combinations may be used to address CDF and Initiator-category-specific targets on redundancy, diversity and functional reliability

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
DRAFT SAFETY EVALUATION REPORT RELATED TO GRAND GULF
EARLY SITE PERMIT APPLICATION
JUNE 2, 2005
ROCKVILLE, MARYLAND

**PROPOSED AGENDA**

| TOPIC | PRESENTER | TIME |
|---|---|---|
| I. Introductory Remarks, ACRS Subcommittee Chairman | Dr. Dana Powers | 8:35- 8:40 am |
| II. SERI Presentation<br>- Overview of Application<br>- Response to NRC issues<br>- Ground motion for GGNP<br>- Seismic issues<br>- Schedule | G. Zinke, et.al. | 8:40- 9:30 am |
| III. NRC Staff Presentation<br>- Review status<br>- Open items<br>- Upcoming Milestones<br>- Schedule | R. Anand, et.al.. | 9:30- 9:55 am |
| IV. General Discussion | | 9:55- 10:00 am |

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

# Presentation to the Advisory Committee on Reactor Safeguards

# Early Site Permit Application for the Grand Gulf Nuclear Station Site

June 2, 2005

Entergy / System Energy Resources, Inc.

Entergy®

Entergy

Item #8, handout

# Entergy ESP Team

- Entergy
  - George Zinke
  - Kenneth Hughey
  - Michael Bourgeois
- Enercon
  - Guy Cesare
- William Lettis & Associates
  - William Lettis

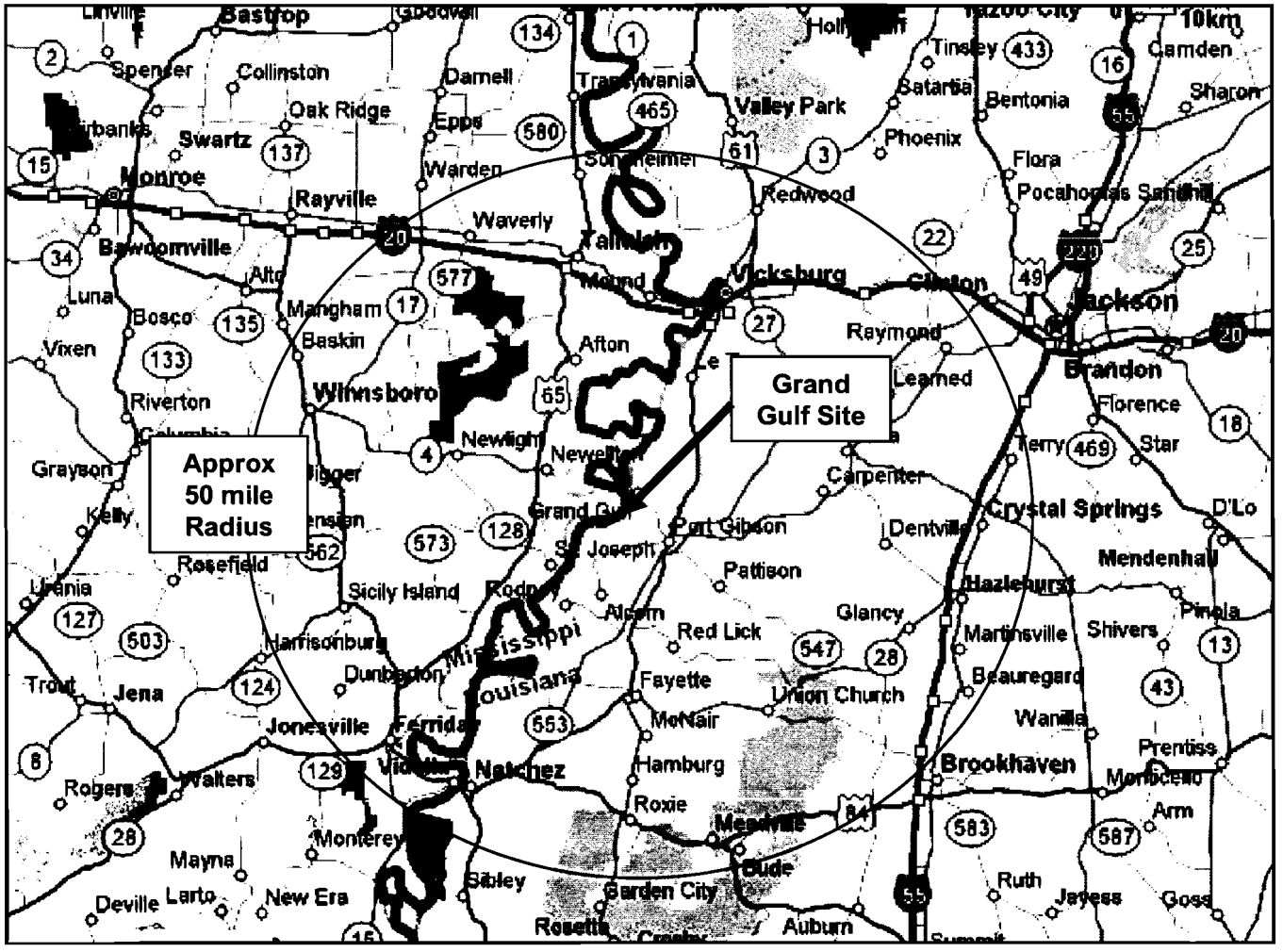
Entergy

# GGNS Site and Environs

- Location: Claiborne County, MS; eastern bank of Mississippi River
- GGNS Site Property - Approx 2100 acres
- Nearest "population center": Vicksburg, MS (approx. 25 miles N; population approx. 27,000)
- Principal town in site vicinity: Port Gibson, MS (approx. 6 miles, SE; population approx. 1750)

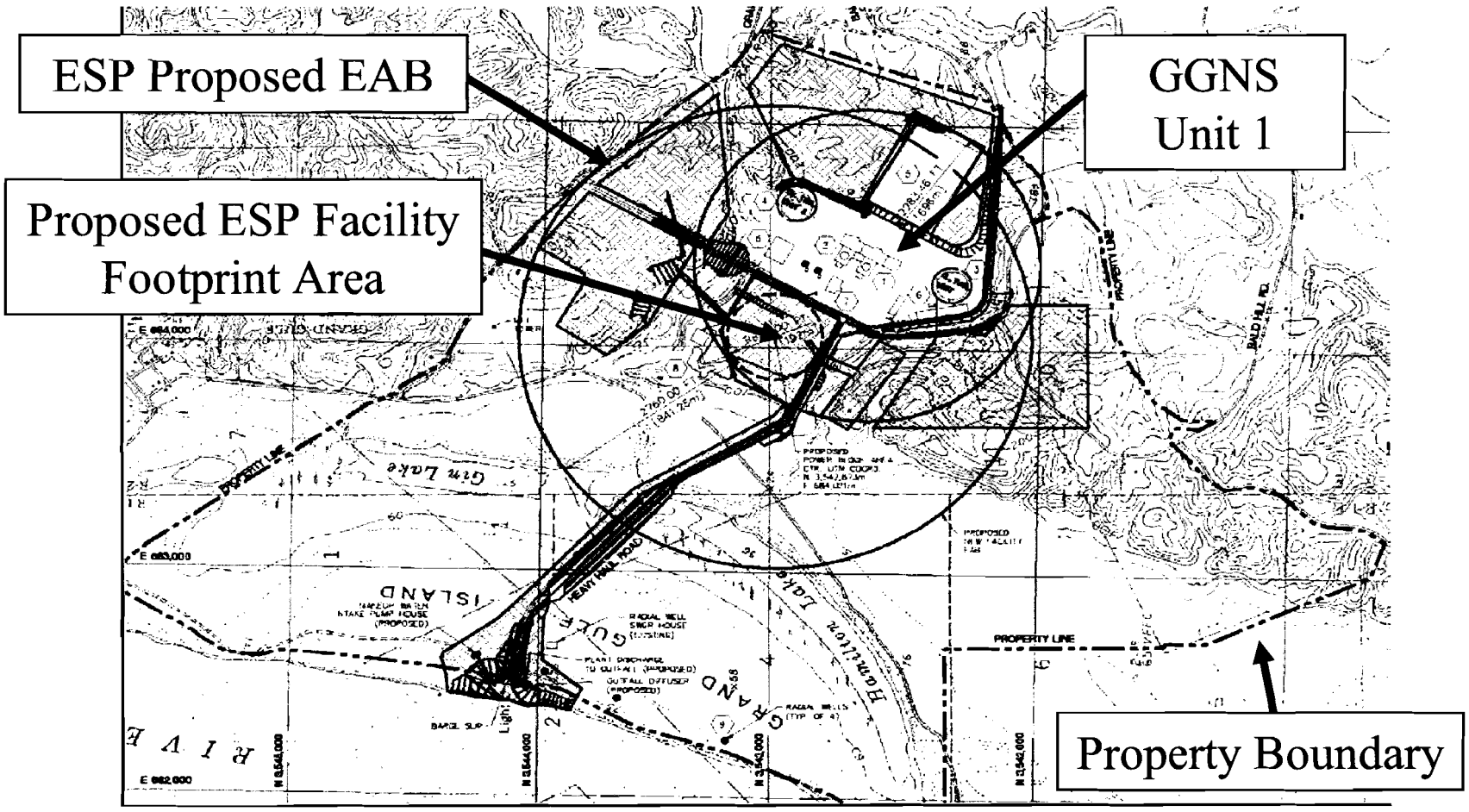
Entergy

# GGNS Site and Environs, Cont'd



Entergy

# GGNS Site and Environs, Cont'd



Entergy

# GGNS Site and Environs, Cont'd

- Site Area Population
  - Permanent (2000 Census)
    - 0 to 10 Miles: 10,000 (approx)
    - 10 to 50 Miles: 325,000 (approx)
- Projections
  - Methodology: MS and LA State
  - Estimated growth, Low to Modest
    - 2030 (Permit Expiration): 8%
    - 2070 (Facility End-of-Life): 19%

Entergy

# GGNS Site and Environs, Cont'd

- Generally rural, remote area; land use, forestry and agriculture; limited industry (primarily lumber)
- No commercial airports with 10 miles; closest at 65 miles (Jackson Intl)
- Closest major highway (US 61), 4.5 miles East of site
  - Minimum safe distance, explosive truck cargo calculated to be 0.3 miles (Unit 1 UFSAR)

Entergy

# GGNS Site and Environs, Cont'd

- No active rail lines or military installations in vicinity
- Closest gas/oil pipeline: 4.75 miles (4", natural gas)
- Current air traffic corridors (commercial and military) evaluated met NRC criteria for no undue risk
- Mississippi River, important river transportation corridor; 1.1 miles from ESP site

Entergy

# GGNS Site and Environs, Cont'd

- At proposed EL 132' (approx), site is located 65' above normal MS River levels.
  - River West bank levee structure: EL 103'
  - River flood height, >29' below site grade
- Consideration of river-borne hazards
  - Updated shipment information considered
  - Distance and river bluff provide protective feature
  - Unit 1 UFSAR analyses, as supplemented, and conclusions remain applicable
- Unit 1 $H_2$ shipments analyzed to be non-hazard

Entergy

# GGNS Site and Environs, Cont'd

- Exclusion area boundary, revised to encompass proposed new facility
  - No residents within EAB; not traversed by rail or navigable waterway; one county road crosses through EAB
  - New EAB, wholly contained with GGNS site property boundary
- Low Population Zone - 2 mile radius (essentially unchanged from Unit 1)

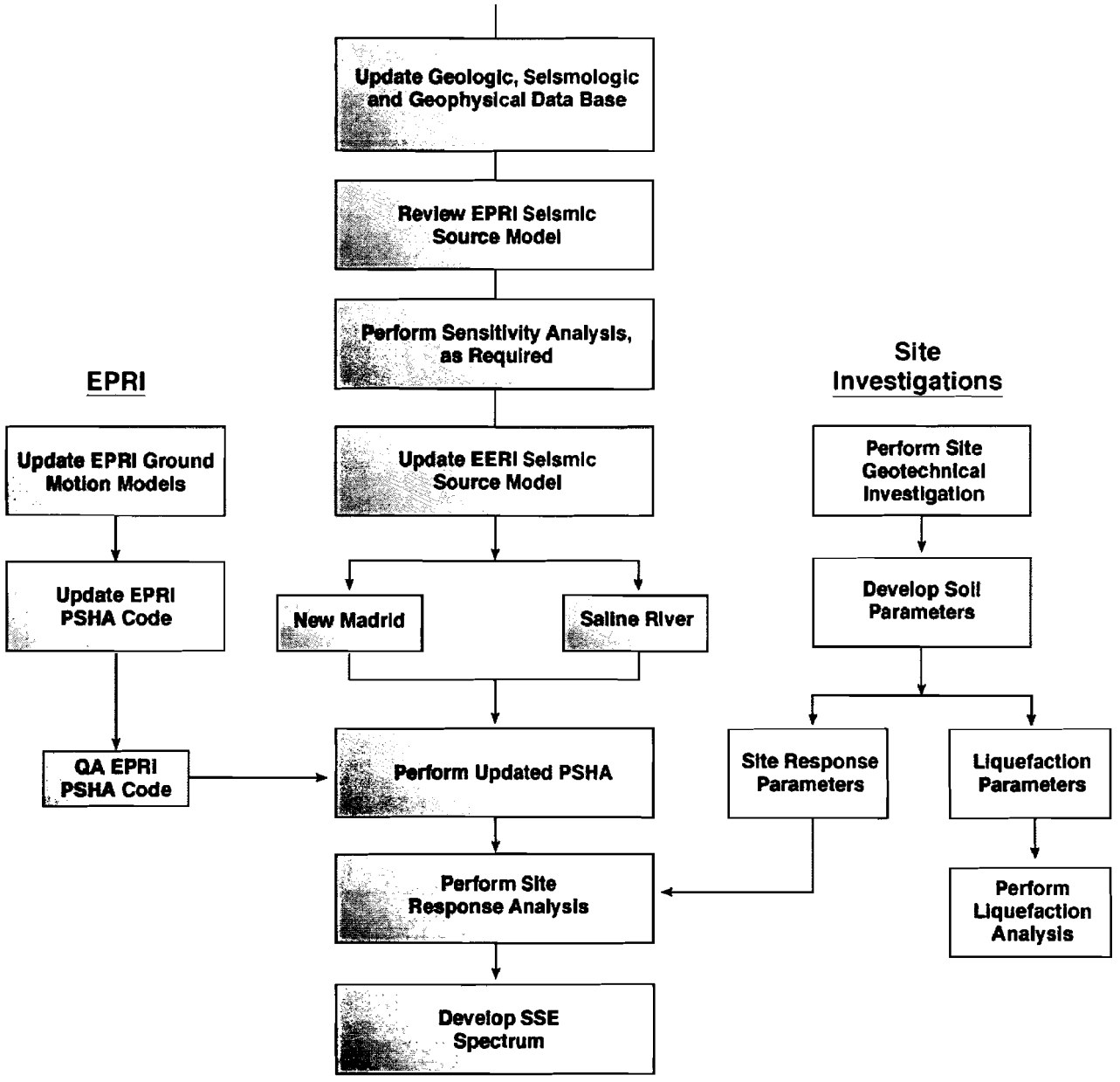
*Geologic and Geotechnical Site Characterization*

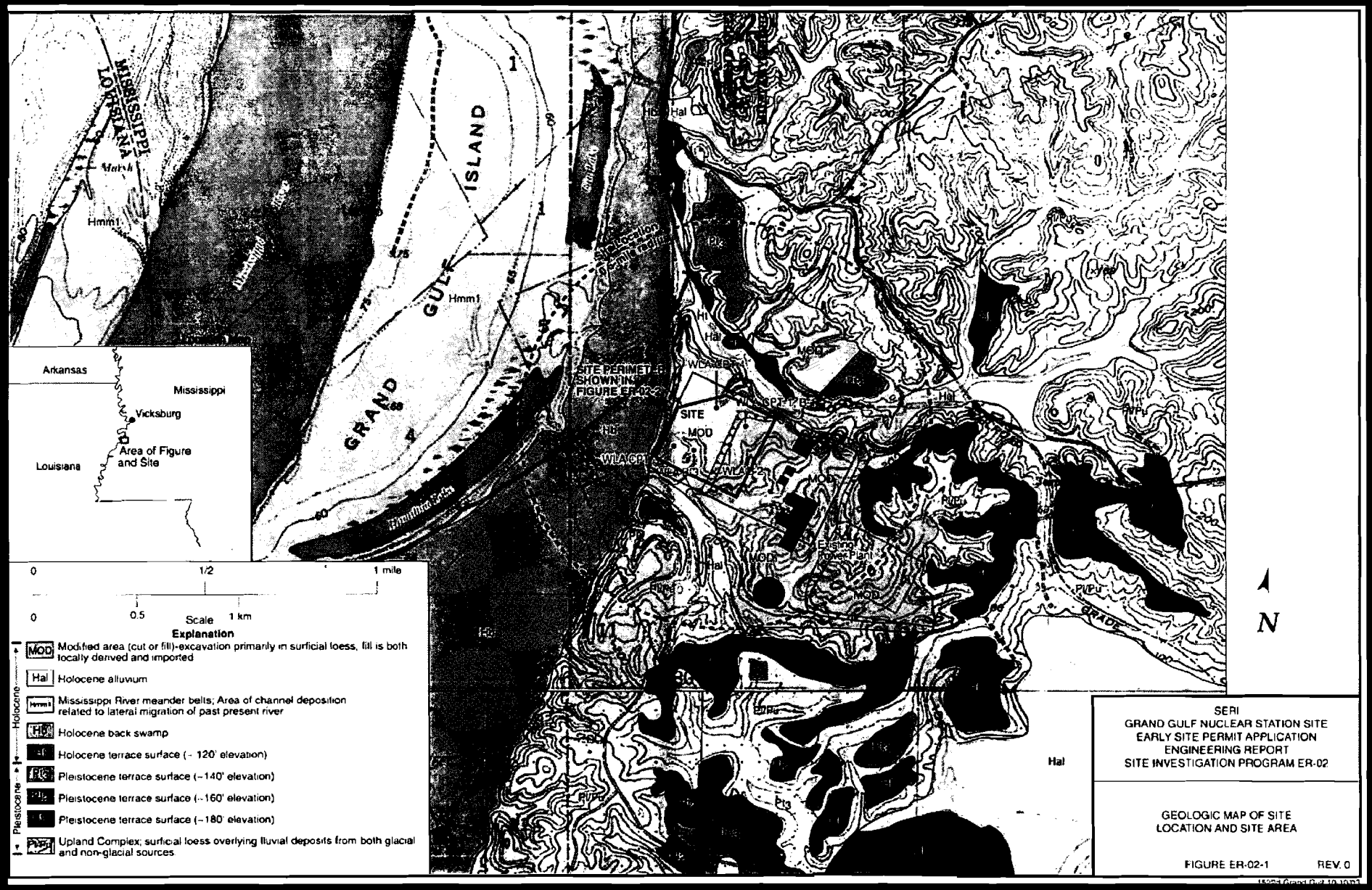
*Regional Geology*

*Seismic Source Characterization*

*PSHA, Site Response and SSE*

# Grand Gulf ESP
# Ground Motion Analysis

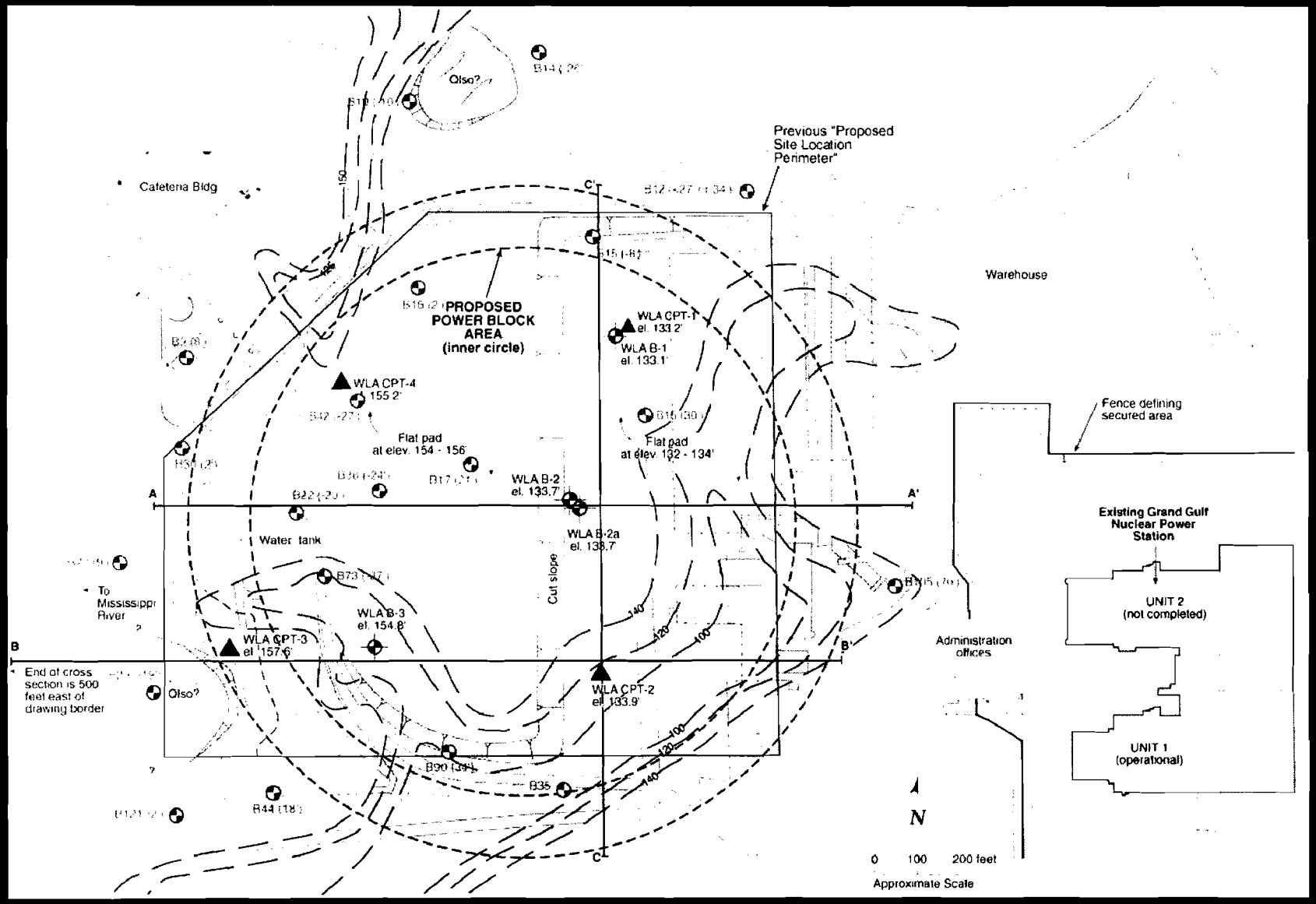


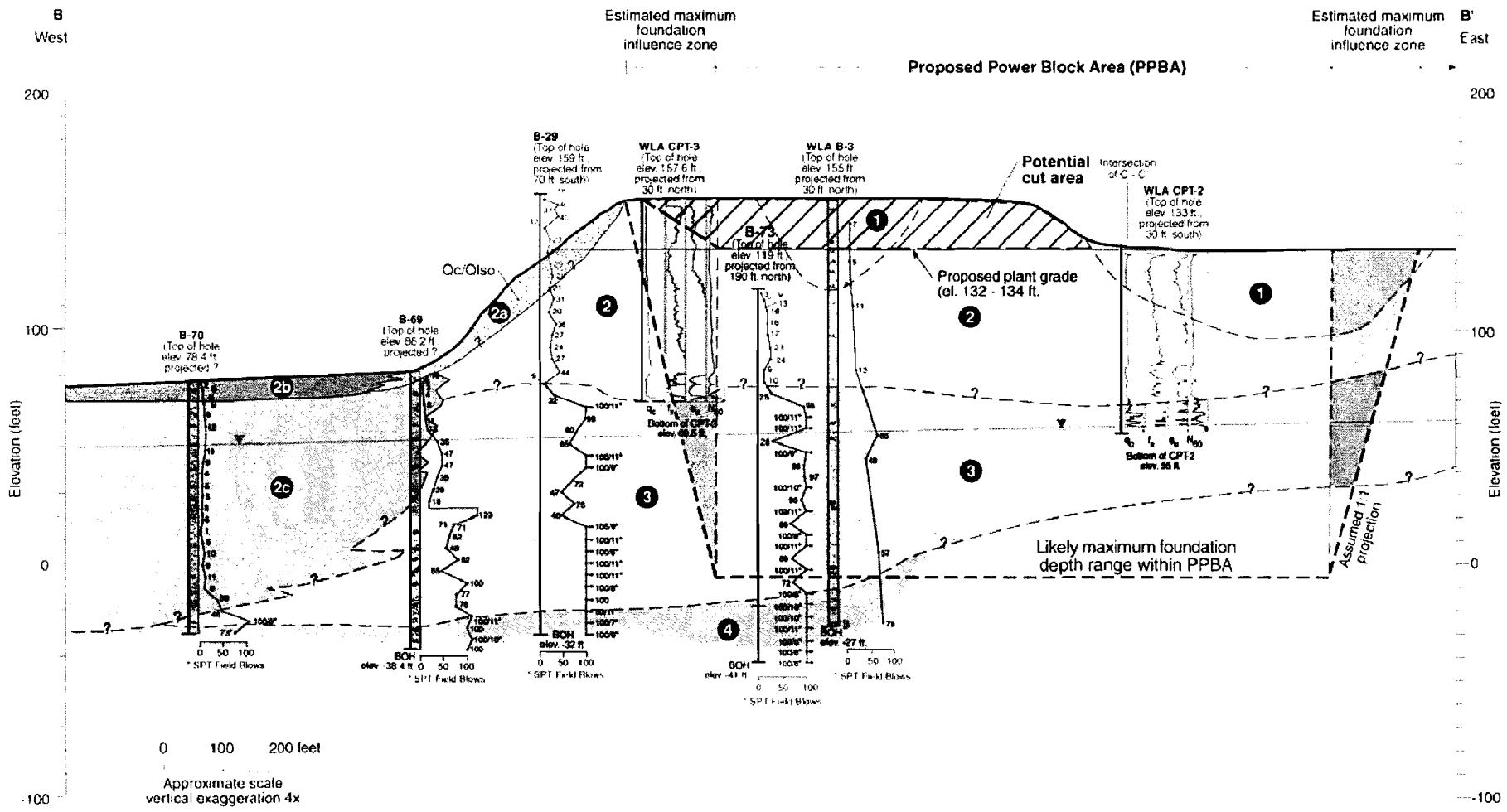


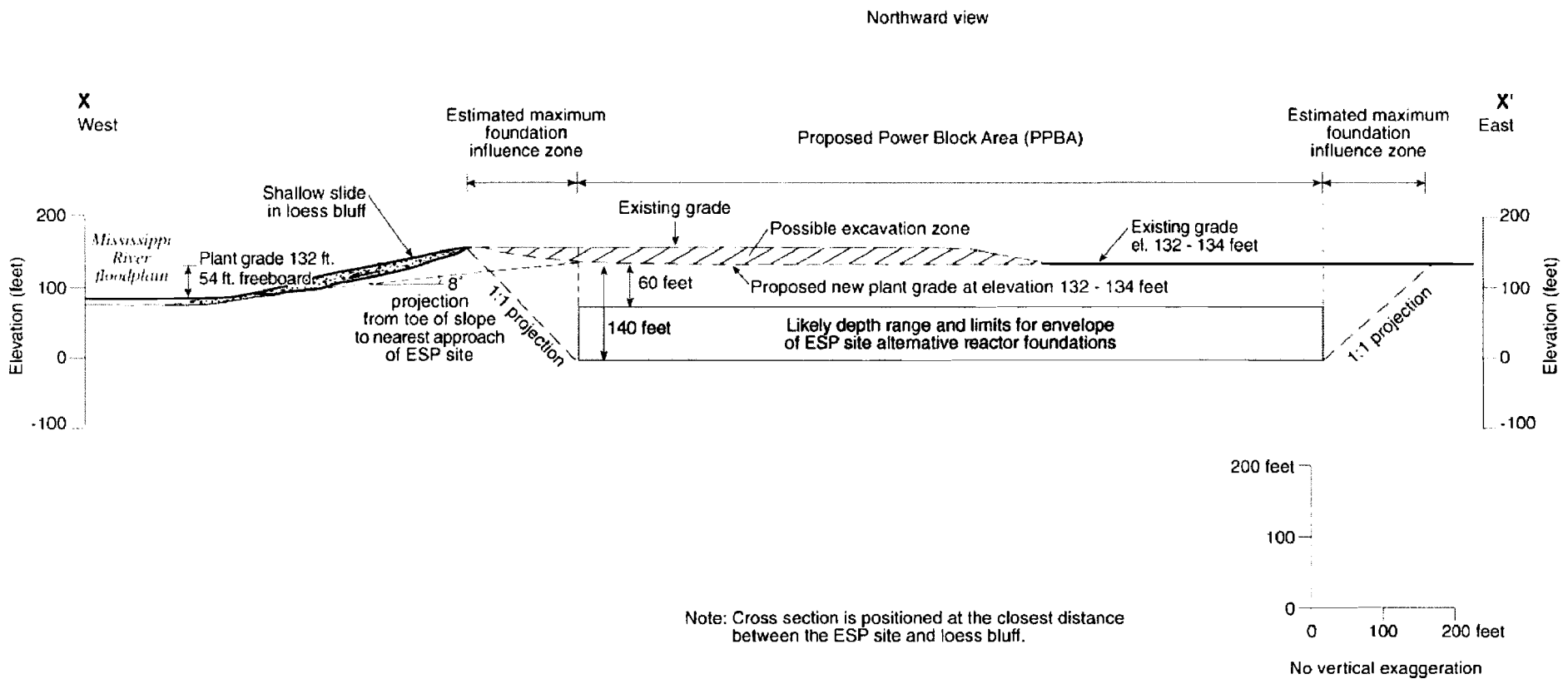

SERI
GRAND GULF NUCLEAR STATION SITE
EARLY SITE PERMIT APPLICATION
ENGINEERING REPORT
SITE INVESTIGATION PROGRAM ER-02

GEOLOGIC MAP OF SITE
LOCATION AND SITE AREA

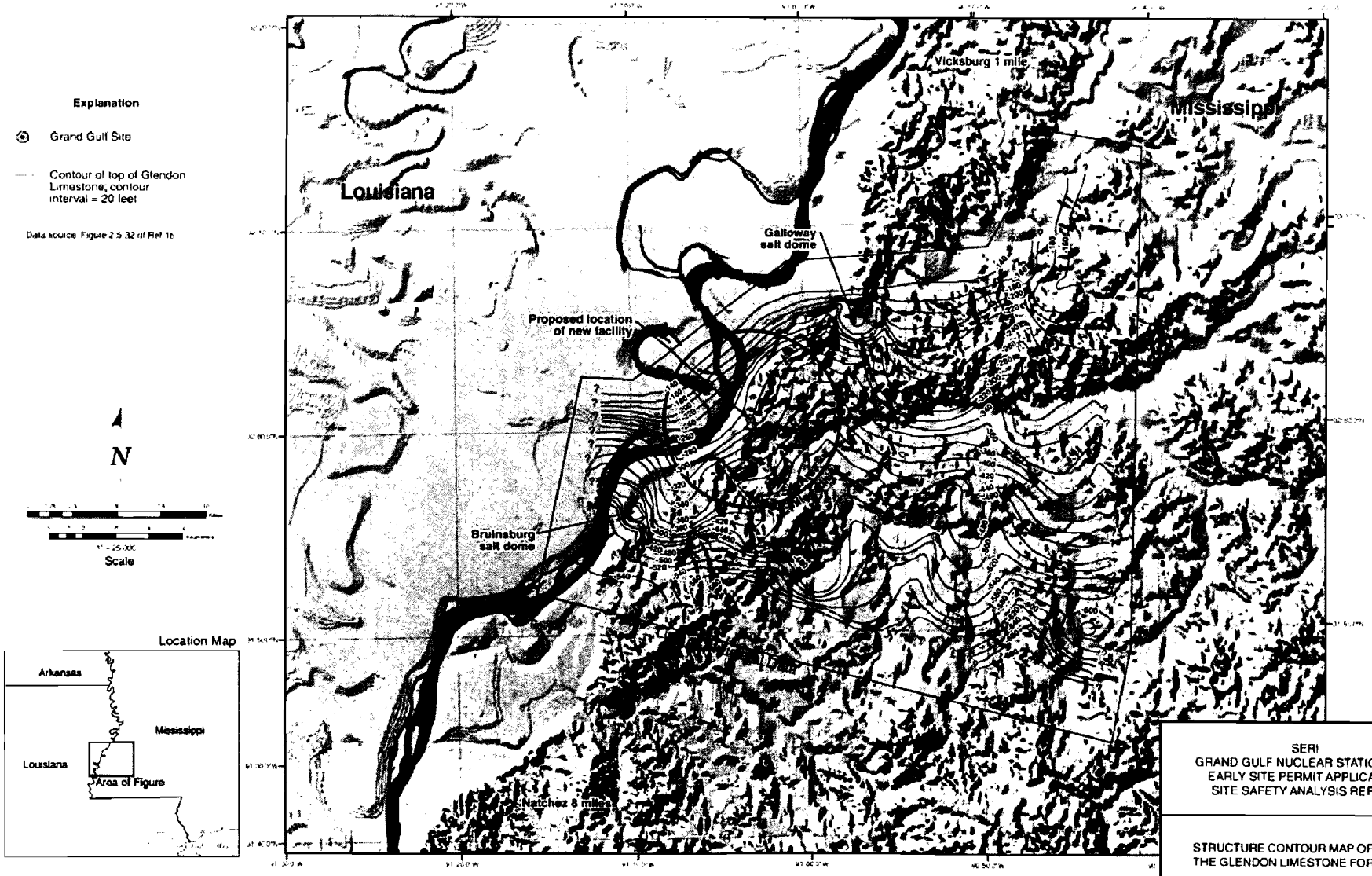
FIGURE ER-02-1 REV. 0


Previous "Proposed Site Location Perimeter"
Cafeteria Bldg
Warehouse
PROPOSED POWER BLOCK AREA (inner circle)
WLA CPT-1 el. 133.2'
WLA B-1 el. 133.1'
WLA CPT-4 155.2'
Flat pad at elev. 154 - 156'
Flat pad at elev. 132 - 134'
WLA B-2 el. 133.7'
WLA B-2a el. 138.7'
Water tank
Cut slope
WLA B-3 el. 154.8'
WLA CPT-3 el. 157.6'
WLA CPT-2 el. 133.9'
To Mississippi River
End of cross section is 500 feet east of drawing border
Olso?
Fence defining secured area
Existing Grand Gulf Nuclear Power Station
UNIT 2 (not completed)
UNIT 1 (operational)
Administration offices
N
0 100 200 feet
Approximate Scale


B
West
Estimated maximum foundation influence zone
Proposed Power Block Area (PPBA)
Estimated maximum foundation influence zone
B'
East
200
100
0
-100
Elevation (feet)
B-29
(Top of hole elev 159 ft projected from 70 ft south)
WLA CPT-3
(Top of hole elev 157.6 ft projected from 30 ft north)
WLA B-3
(Top of hole elev 155 ft projected from 30 ft north)
Potential cut area
Intersection of C - C'
WLA CPT-2
(Top of hole elev 133 ft projected from 30 ft south)
B-73
(Top of hole elev 119 ft projected from 190 ft north)
Proposed plant grade (el. 132 - 134 ft.
Qc/Qlso
B-69
(Top of hole elev 86.2 ft projected ?
B-70
(Top of hole elev 78.4 ft projected ?
Likely maximum foundation depth range within PPBA
Assumed 1:1 projection
Bottom of CPT-3 elev. 60.5 ft.
Bottom of CPT-2 elev. 56 ft.
BOH elev. -38.4 ft
BOH elev. -32 ft
BOH elev. -41 ft
BOH elev. -27 ft.
* SPT Field Blows
0 50 100
0 100 200 feet
Approximate scale
vertical exaggeration 4x



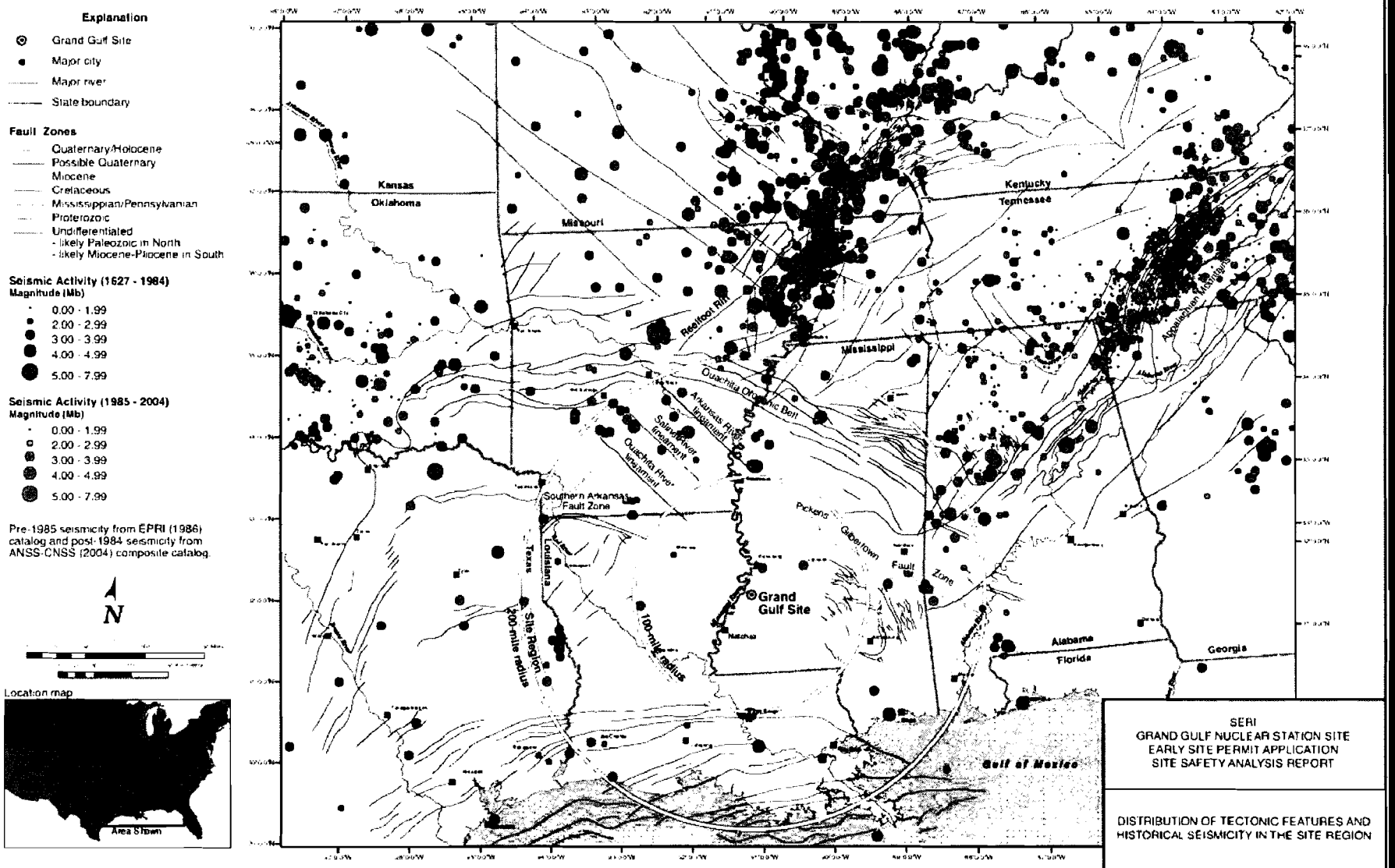

Note: Cross section is positioned at the closest distance between the ESP site and loess bluff.




SERI
GRAND GULF NUCLEAR STATION SITE
EARLY SITE PERMIT APPLICATION
SITE SAFETY ANALYSIS REPORT

STRUCTURE CONTOUR MAP OF TOP OF THE GLENDON LIMESTONE FORMATION

FIGURE 2.5-15 REV. 0

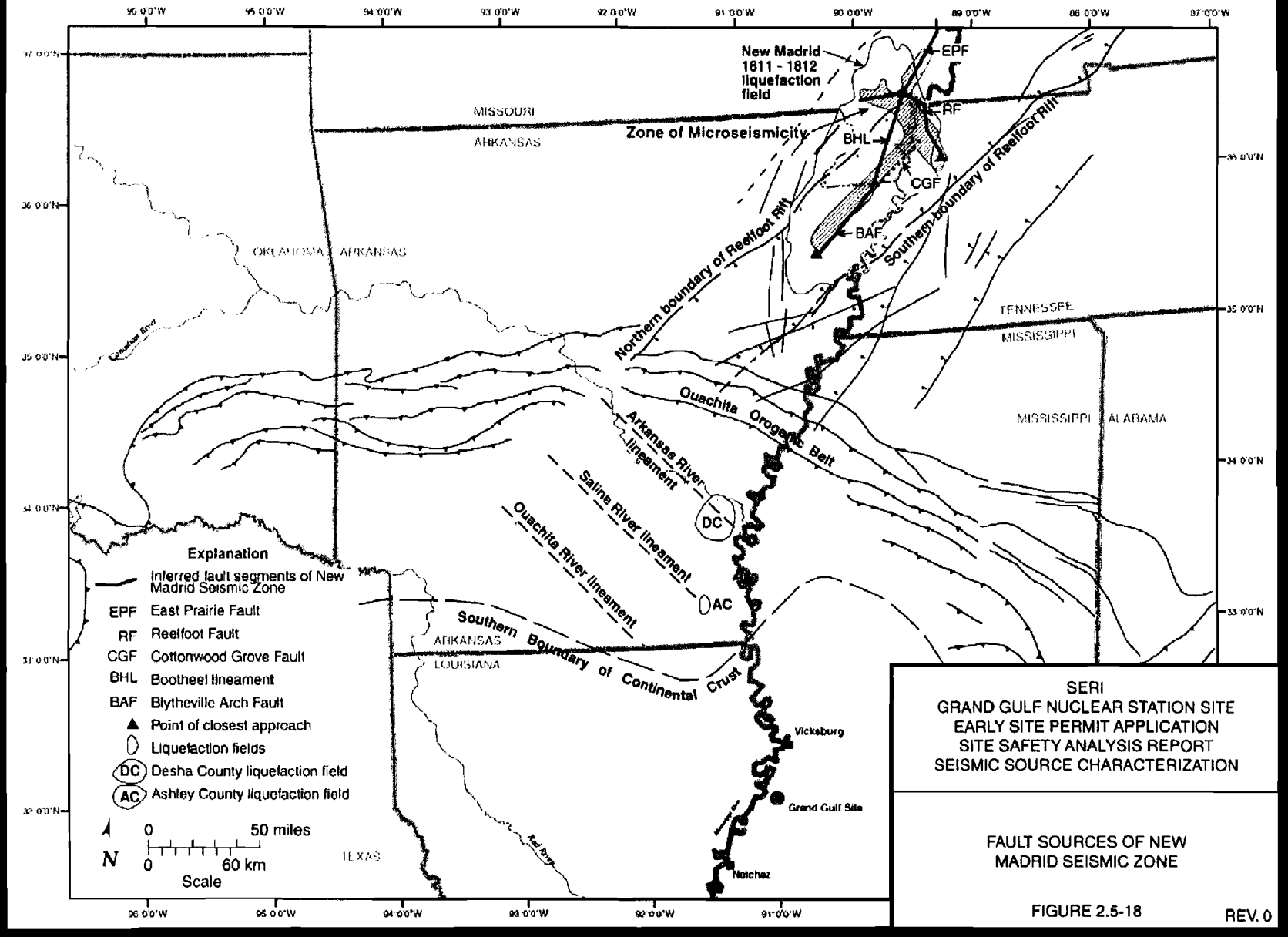



SERI
GRAND GULF NUCLEAR STATION SITE
EARLY SITE PERMIT APPLICATION
SITE SAFETY ANALYSIS REPORT

DISTRIBUTION OF TECTONIC FEATURES AND HISTORICAL SEISMICITY IN THE SITE REGION

02/11/05 FIGURE 2.5-5 DRAFT REV. 1

1322 SSAR Rev 1




SERI
GRAND GULF NUCLEAR STATION SITE
EARLY SITE PERMIT APPLICATION
SITE SAFETY ANALYSIS REPORT
SEISMIC SOURCE CHARACTERIZATION

FAULT SOURCES OF NEW MADRID SEISMIC ZONE

FIGURE 2.5-18

REV. 0

GGNS SEISMIC HAZARD RESULTS FOR
$S_a$(5 Hz) FOR ROCK SITE CONDITIONS

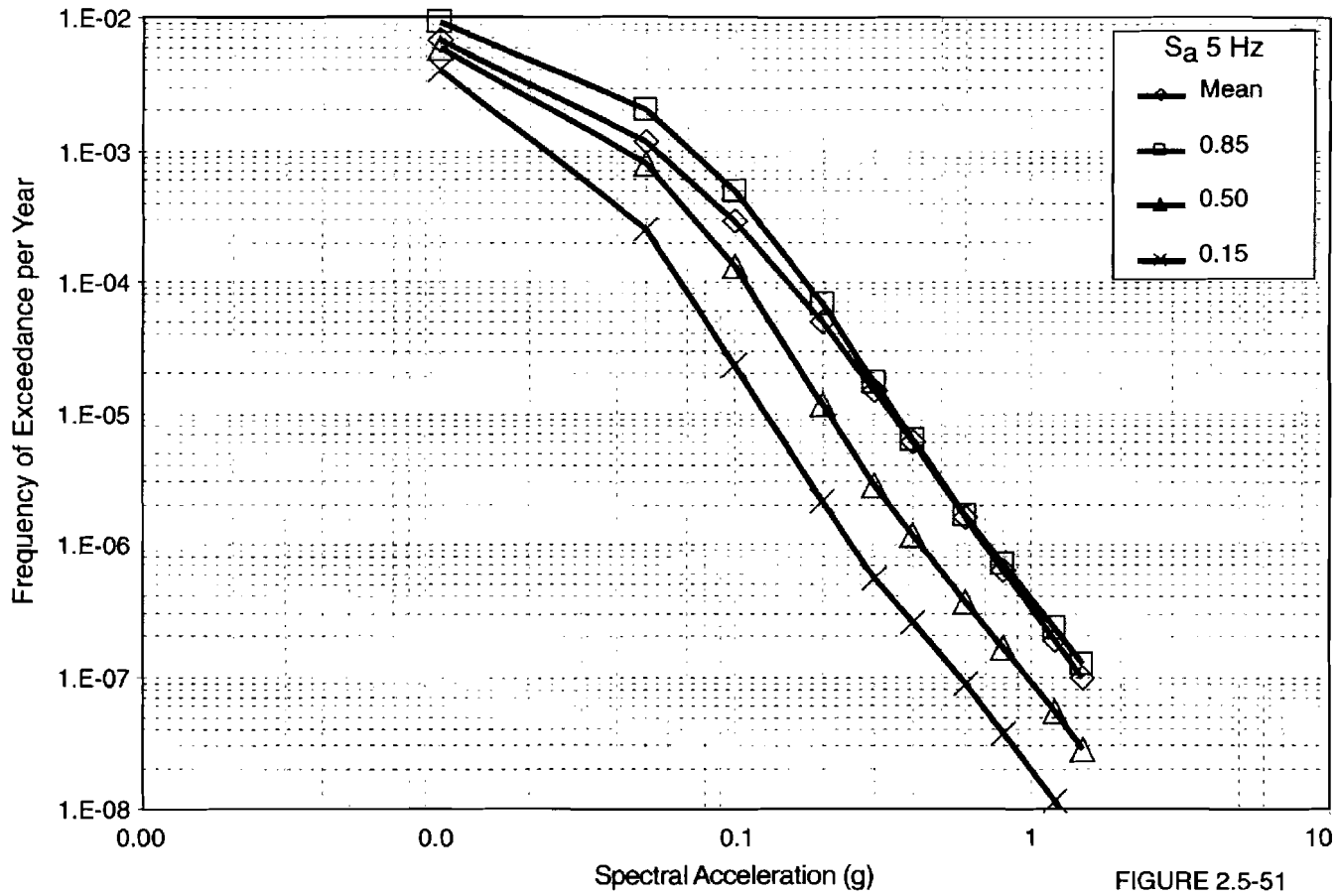



FIGURE 2.5-51

MEAN TRANSFER FUNCTIONS
CORRESPONDING TO 1 TO 2 HZ AND 5 TO
10 HZ SCALED SPECTRA (FIGURE 63) AND
ENVELOP: TOP OF LOESS

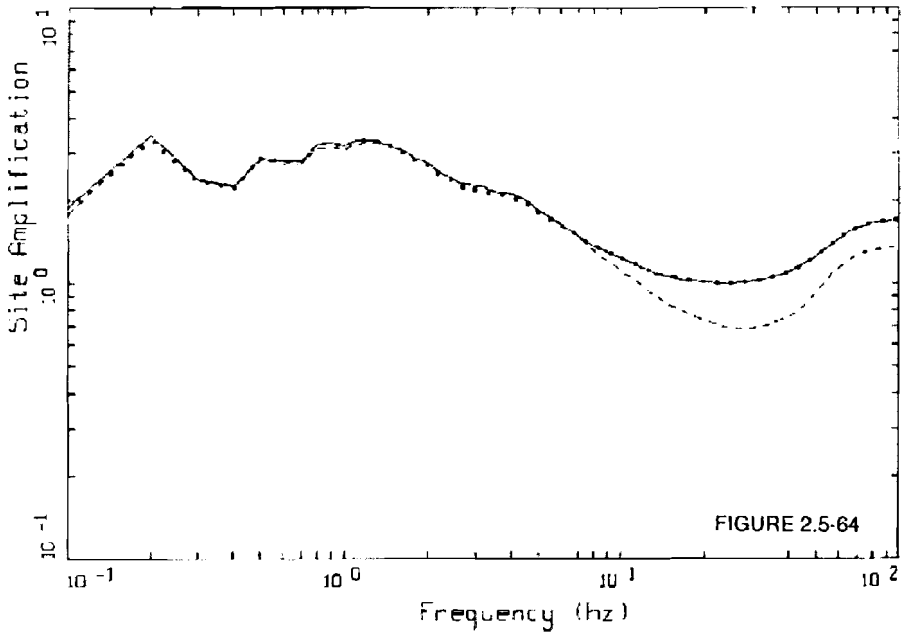
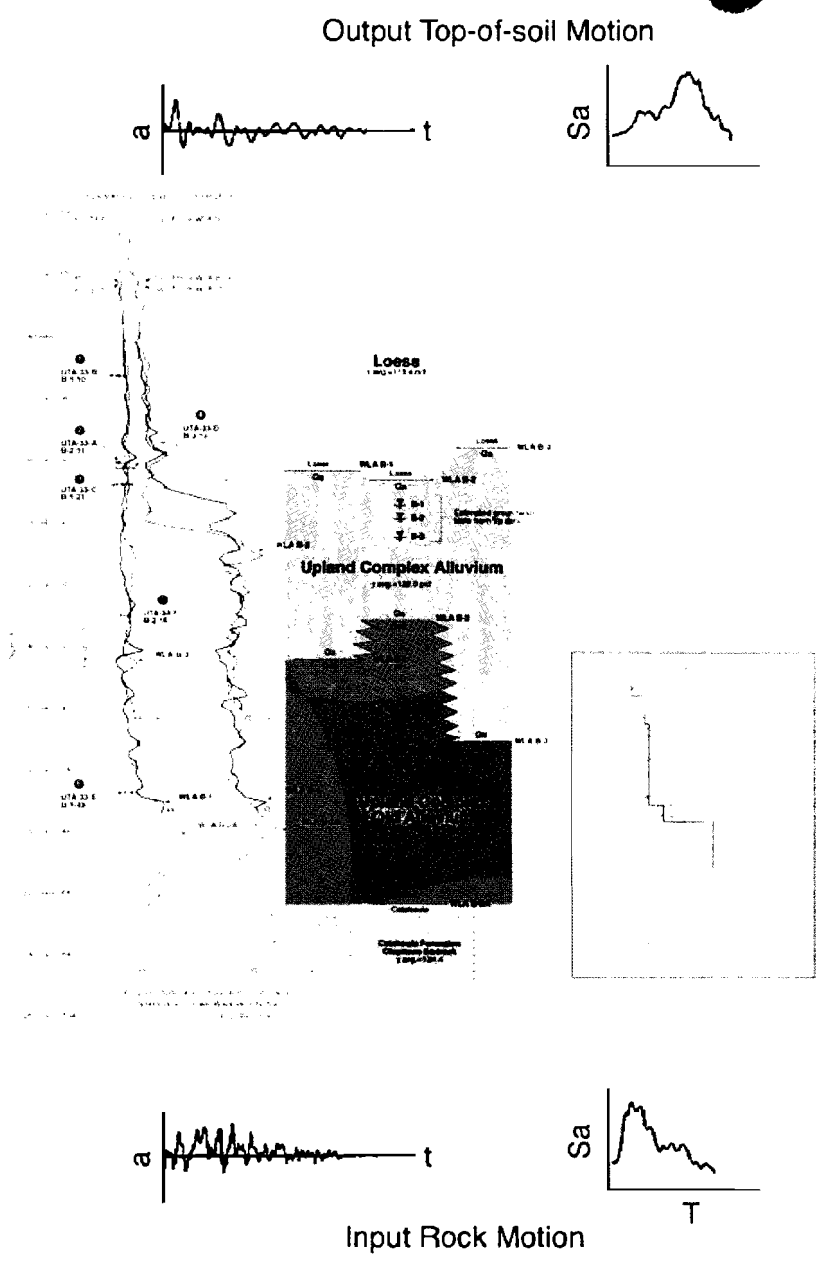
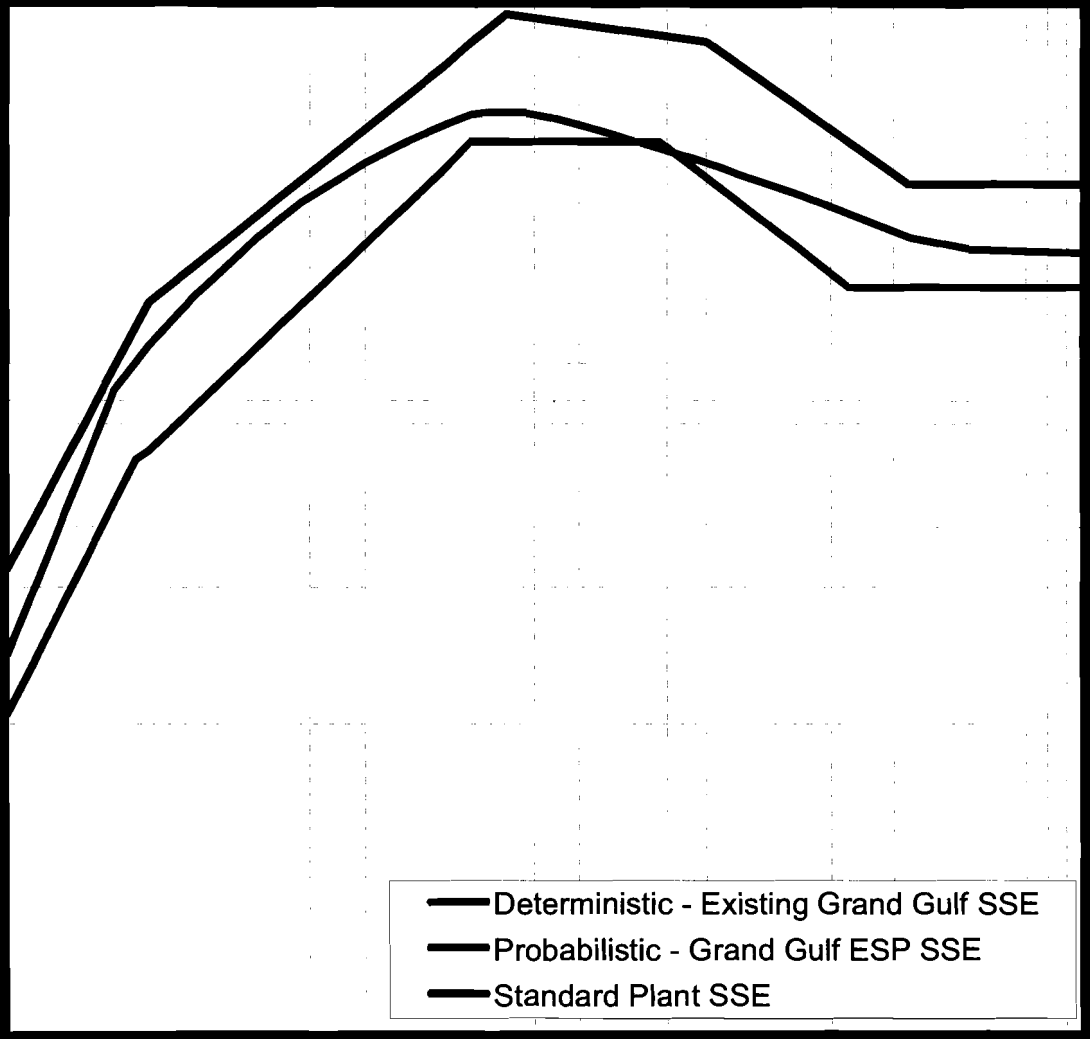



FIGURE 2.5-64

TOP OF LOESS
MEAN AMPLIFICATION

LEGEND
——— ENVELOPE
• • • • MEAN 1-2 HZ
- - - - - MEAN 5-10 HZ




Deterministic - Existing Grand Gulf SSE
Probabilistic - Grand Gulf ESP SSE
Standard Plant SSE

Entergy

# DSER Open Items

- 23 Open Items
- Responses due June 21
- Status matrix attached

Entergy

# CONCLUSIONS

- Site characterized in accordance with 10CFR52 and 10CFR100
- Site acceptable for nuclear plant(s) which fall within the site characteristics
- No physical characteristics pose a significant impediment to development of emergency plans

# DSER OPEN ITEM MATRIX

| OI No. | DSER Section | Subject | DSER Open Item Proposed Resolution |
|---|---|---|---|
| 2.1-1 | 2.1.2.3 | Demonstrate that the applicant has control over the exclusion area or has a right to obtain such control. | Unknown; need feedback from OGC |
| 2.1-2 | 2.1.3.3 | Include weighted transient population data in Tables 2.1-1 and 2.1-2 of the SSAR. | Weighted transient population will be included in the SSAR for distances from 10 miles out to 30 miles from the ESP site; Tables 2.1-1 and 2.1-2 of the SSAR will remain as is for "permanent" resident population data.<br>The 0 – 10 mile transient population distribution is provided in the ESP application Part 4 EPI section, and will not be revised. The response will provide information sufficient to show compliance with Part 100 population center distance criteria and RG 4.7 population density (criteria applies, explicitly for areas out to 20 miles, as noted in DSER). |
| 2.3-1 | 2.3.1.3 | Provide acceptable 100-year return period maximum and minimum dry-bulb temperatures. | Conservative 100-year return maximum and minimum temperatures will be provided in SSAR Section 2.3.2.1.2. Additional information will be provided regarding applicability and use of databases beyond those cited in the SSAR application (and/or previous RAI responses). |
| 2.3-2 | 2.3.1.3 | Provide the 48-hour probable maximum winter precipitation (PMWP) that can be used with the 100-year snowpack to define the extreme winter precipitation load site characteristics. | The SSAR Section 2.3.1.2.5 will be revised to provide a conservative value for a 48-hour PMWP to be used with the 100-year return snowpack (Section 2.3.1.2.4) to determine extreme winter precipitation design loads (Section 2.3.1.2.6). |
| 2.3-3 | 2.3.1.3 | Identify an additional ultimate heat sink (UHS) meteorological site characteristic for use in evaluating the potential for water to freeze in the UHS water storage facility. | Response will provide a site characteristic in terms of accumulated freezing degree days (AFDD), which is an integration of (32 – temperature) over time in units of days as described in US Army Corp of Engineer report ERDC/CRREL Technical Note (TN) 04-3, June 2004, based on 72 yrs of data. |
| 2.3-4 | 2.3.1.3 | Identify a 3-second gust wind speed that represents a 100-year return period for the ESP site. | Response will specify a 3-second gust wind speed as a site characteristic based on SEI/ASCE 7-02, "Minimum Design Loads for Buildings and Other Structures", 2002. |
| 2.3-5 | 2.3.5.3 | Identify x/Q and D/Q values for the nearest milk cow and meat cow. | Atmospheric dispersion and diffusion coefficients for the nearest milk and meat cows will be identified in the response, and Table 2.3-143 of the SSAR and Table 2.7-117 of the ER will be revised to include the parameters. |

| OI No. | DSER Section | Subject | DSER Open Item Proposed Resolution |
|---|---|---|---|
| 2.4-1 | 2.4.1 | Provide corrected UTM coordinates of the center of the proposed power block and/or revise Figure 2.1-1 in the SSAR to show the correct location and coordinates. | Corrected UTM coordinates (UTM NAD83, Zone 15 – 96W to 90W, meters) of N3,543,261 meters and E684,018 meters will be indicated in the response, and will be shown on relevant drawings and in the text of the SSAR, ER and EPI parts of the application. |
| 2.4-2 | 2.4.1 | Provide information on the elevation (depth) of the zone that could be disturbed by the construction of the new facility, such that the local subsurface environment and its alignment with the existing hydrogeological environment could be altered. | Response will provide discussion of the existing SSAR text (as supplemented by revisions due to RAI responses) with regard to the depth of excavation required for a new facility. Response will reiterate that the specific plant design has not been identified, therefore, details of excavation depths cannot be provided. This is a COL item. The response will also discuss in general terms the potential impact to the local subsurface hydrogeologic environment. |
| 2.4-3 | 2.4.1 | Provide more details regarding dewatering wells to allow the staff to determine whether ground surface subsidence could affect safety-related structures and piping. Provide information related to the location of dewatering wells in relation to safety-related structures and associated monitoring of the ground water table. | Response will discuss existing site ground water conditions and the potential (likely) need for dewatering during construction. Reference to discussions if SSAR Section 2.5 will be made regarding soil characteristics and subsidence potential. Discussion of a ground water monitoring program for the pre-construction and construction periods will be included (this is also relevant to Open Items 2.4-2 and 2.4-7). The discussion of plant water use during operation, and its potential impact on ground surface subsidence will also be discussed. Specific locations of dewatering wells in relation to safety related structures is a design feature, and will be the responsibility of the COL applicant. |
| 2.4-4 | 2.4.1 | Provide more details regarding the floodwater level estimation, including data and methods used to arrive at the floodwater elevation of 133.25 feet MSL. | The response will discuss the need to evaluate local intense precipitation during the COL phase, using rainfall rates from HMR 52, when the plant design and location on the ESP site are known, and site final grade and grading and drainage system designs are completed. Additional assessment of general ESP site drainage features will be provided to qualitatively treat impact of PMP determined per HMR 52 on ESP site. The floodwater estimation in the SSAR which includes an estimated flood elevation of 133.25 ft msl will be annotated to indicate its applicability only to the GGNS Unit 1 plant area and plant |

| OI No. | DSER Section | Subject | DSER Open Item Proposed Resolution |
|---|---|---|---|
| | | | design. |
| 2.4-5 | 2.4.2 | Revise and present estimates of the local intense precipitation as shown in Table 2.4-7 of the SSAR using the guidelines of HMR 52. | Response will provide revised local intense precipitation values for the site based on the methodology of HMR 52. |
| 2.4-6 | 2.4.13 | Provide further description of the rationale for considering Sr-90 and Cs-137 in the radionuclide transport analysis. | Conservative screening analysis of postulated leak, assuming no retention or absorption in soil media, will be used to identify "key nuclides of interest." In response to OI 2.4-7, this revised (likely expanded) listing of nuclides will be then be used to define site characteristic distribution coefficients that would be considered in a revised transport analysis by the COL applicant. |
| 2.4-7 | 2.4-13 | Factors, such as soil, sediment, and rock characteristics; adsorption and retention coefficients; ground water velocity; and distances to the nearest body of surface water are important to hydrological radionuclide transport. Provide these site characteristics from onsite measurements. | The response will indicate that ESP site specific soil samples were used to define the soil types. Soil borings were taken (during original site characterization) defining key soil types from the site to the major water body of interest (i.e., MS River). Response will then define site characteristic Kd values that are appropriate for the listing of "nuclides of interest" (developed in response to OI 2.4-6). |
| 2.5-1 | 2.5.2 | Provide justification for not updating the background seismic source for the ESP site. | Additional information will be provided explaining why the current EPRI Model was acceptable for the GGNS site |
| 2.5-2 | 2.5.2 | Provide and evaluate the criteria or weights used for ranking of model clusters and the judgements involved in balancing data consistency and adherence to seismological principles in the EPRI 2003 ground motion evaluation. Explain how recordings from a single earthquake can provide well-resolved values of both crustal quality factor (Q) and site kappa, also explain why the Q value of 317 at 1 Hz is much lower than values found in other studies of eastern North American earthquakes, and why other studies find less frequency dependence of Q in the eastern North American than in the western North American. | Generic issue for ESP applicants. Will reference Dominion response including docketing the final EPRI Report. |
| 2.5-3 | 2.5.2 | Provide an explanation why the magnitude and distance bin corresponding to the SRSZ makes no contribution to the hazard deaggregation. | Explanation to be provided in addition to referencing generic response. |

| OI No. | DSER Section | Subject | DSER Open Item Proposed Resolution |
|---|---|---|---|
| 2.5-4 | 2.5.2 & 2.5.4 | Provide justification on applying the generic shear wave velocity profile derived from Memphis area to the ESP site and on its applying kappa value derived from ground motion observation on the Mississippi embayment in the sensitivity test. | Comparison of ESP stratigraphic profile to Memphis area to be provided. Description of tbe importance of up to 1-2 kilometers of soil profile to be provided. |
| 2.5-5 | 2.5.4 | Provide the basis for the selection of values of BE, UB, and LB and other parameters for the base case profile. | Will provide basis for selection of strain-compatible soil properties. Soil property already provided in RAI response |
| 13.3-1 | | Provide responses to the following issues related to State and local emergency plans: | NA |
| a | 13.3.3.7 | Describe the communications arrangements with fixed and mobile medical support for the State of Mississippi and with mobile medical support for Claiborne County. | No new information to be provided; resolve at COL |
| b | 13.3.3.8 | Describe the dissemination of information regarding the special needs of the handicapped to the general public in the State of Louisiana on a periodic basis. | No new information to be provided; resolve at COL |
| c | 13.3.3.11 | Describe the means for the use of radioprotective drugs for emergency workers and institutionalized persons within the plume exposure pathway EPZ in the States of Louisiana and Mississippi whose immediate evacuation may be infeasible or very difficult. | No new information to be provided; resolve at COL |
| d | 13.3.3.12 | Describe the State of Mississippi's guidance related to bioassay or whole body counting for determining offsite emergency worker doses from the uptake of radioactive material (e.g., ingestion) | No new information to be provided; resolve at COL |
| e | 13.3.3.13 | Clarify the apparent inconsistencies between the LPRRP and Enclosure I to Attachment 2 to LPRRP Supplement II regarding the description of contacts and arrangements for local and backup hospital services. | No new information to be provided; resolve at COL |
| f | 13.3.3.13 | Describe the special radiological capabilities for the hospitals listed in Tab 2 of LPRRP Chapter 10. | No new information to be provided; resolve at COL |
| g | 13.3.3.11 | Provide information regarding the availability and capacity of school buses or other transportation methods, the availability of drivers, and the process for mobilizing transportation for students, residents, | No new information to be provided; resolve at COL |

| OI No. | DSER Section | Subject | DSER Open Item Proposed Resolution |
|---|---|---|---|
| | | transients, and special needs populations in Claiborne County and Tensas Parish during an evacuation (e.g., evacuations may require a single trip or they may require return trips). | |
| h | 13.3.3.11 | Provide a map(s) illustrating evacuation/shelter areas in the State of Mississippi for the MREPP Annex O. | No new information to be provided; resolve at COL |
| i | 13.3.3.11 | Information on shelter capacities is not contained in, and therefore, not evaluated by FEMA under the LPRRP. Provide sheltering capacities for relocation centers in the State of Louisiana or documentation of evaluation performed to determine whether adequate capacity exists. | No new information to be provided; resolve at COL |
| 13.3-2 | 13.3.3.8 | Describe in Part 4 the applicant's responsibility for making information available to offsite authorities for distribution consistent with MREPP Annex J. | No new information to be provided; resolve at COL |
| 13.3-3 | 13.3.3.9 | Describe the adequacy of the TSC, OSC, and EOF and related equipment used to support emergency response activities, to address, with specificity, such facility and equipment features as location, size, structure, habitability, communications, staffing and training, radiation monitoring, instrumentation, data system equipment, power supplies, technical data and data systems, and record availability and management. | No new information to be provided; resolve at COL |
| 13.3-4 | 13.3.3.11 | Address whether discussions on results of the 2003 ETE study were held with officials from the States of Mississippi and Louisiana involved in implementing traffic management plans, according to Appendix 4 to NUREG-0654/FEMA-REP-1 and NUREG/CR-4831, or provide confirmation that State reviews were not required based on discussions with appropriate officials. | No new information to be provided; resolve at COL |

Item #8, handout 2

# Presentation to the Advisory Committee on Reactor Safeguards

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# Safety Review of the Grand Gulf Early Site Permit Application

**Presented by**
**Raj Anand**
**Project Manager**
**New, Research and Test Reactors Program**
**June 2, 2005**

# Purpose

- Brief the Committee on the Grand Gulf early site permit (ESP) application and the status of the NRC staff's safety review of that application

- Support the Committee's review of the application and subsequent interim letter from ACRS to the EDO

- Answer the Committee's questions

# Agenda

| | |
|---|---|
| • Background and Regulatory Framework | 1 min |
| • Completed Milestones | 1 min |
| • Summary of Open Items | 1 min |
| • DSER Conclusions | 1 min |
| • Future Milestones | 1 min |
| • Grand Gulf ESP Application | 5 min |
| • Presentation Conclusions | 1 min |
| • Discussion / Committee questions | 5 min |

# Background and Regulatory Framework

- Subpart A to 10 CFR Part 52 governs ESPs
- Subpart B to 10 CFR Part 100 contains applicable siting evaluation factors
- 10 CFR 52.23 requires ACRS to report to Commission on portions of application that pertain to safety (i.e., Site Safety Analysis Report)
- Grand Gulf is third of three ESP applications the NRC staff is currently reviewing – First is North Anna, Second is Clinton, and Third is Grand Gulf

# Completed Milestones

- System Energy Resources (SERI) submitted ESP application by on 10/16/2003
- Staff docketed application on 11/21/2003
- Staff issued draft safety evaluation report (DSER) on 4/7/2005
- Staff issued draft environmental impact statement on 4/21/2005
- Staff & Applicant briefed Subcommittee on 5/16/2005

# Safety Review Areas and Lead Staff Reviewers

- Meteorology: Brad Harvey
- Hydrology: Goutam Bagchi (contract support from Pacific Northwest Laboratory) (PNNL)
- Site Hazards: Kaz Campe (contract support from PNNL)
- Geology/seismology: Yong Li (support from U.S. Geologic Survey and BNL)
- Demography/Geography: Jay Lee
- Emergency Planning: Joe Anderson (consultation with Federal Emergency Management Agency)
- Quality Assurance: Paul Prescott
- Physical Security: Al Tardiff
- Radiological Consequence Analysis: Jay Lee

# Summary of Open Items

There are 23 open items in the DSER

- Exclusion Area Authority and control (1)
- Population Distribution (1)
- Meteorology (5)
- Hydrology (7)
- Seismology and Geology (5)
- Emergency Planning (4)

# DSER Conclusions

- DSER defers general regulatory conclusion regarding site safety and suitability to FSER after open items addressed
- Some conclusions from individual sections without open items
  - Applicant has provided appropriate quality assurance measures equivalent to those in 10 CFR Part 50 Appendix B
  - Site characteristics are such that adequate security plans and measures can be developed

# DSER Conclusions

- Additional conclusions from individual sections without open items
  - Applicant has established appropriate atmospheric dispersion characteristics to support design basis radiological calculations
  - Based on PPE and site characteristics, site meets radiological dose consequence criteria in 10 CFR 50.34(a)(1)
  - Potential hazards associated with nearby transportation routes, industrial and military facilities pose no undue risk to facility that might be constructed on the site

# Grand Gulf ESP Application

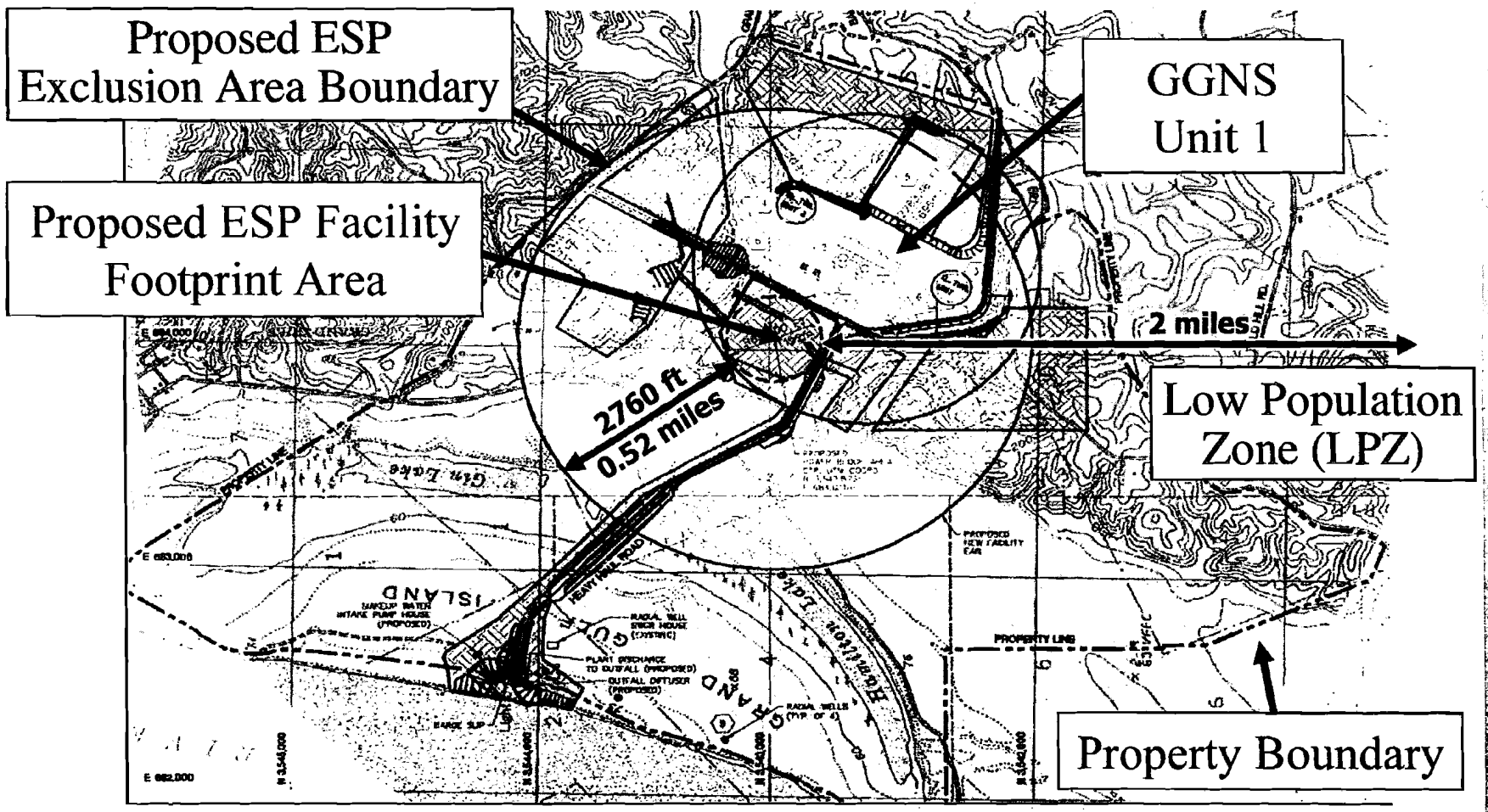
- SERI requests ESP site be approved for total nuclear generating capacity of up to 8600 MWt, with max 4300 MWt per unit
- Each unit may be one large reactor or multiple smaller reactors
- SERI has chosen not to submit a specific design but instead has submitted a plant parameter envelope (PPE) based on a number of current and future reactor designs
- PPE values are selected by the applicant. Staff's review of PPE is limited to whether they are reasonable

# Grand Gulf ESP Application

- Original Grand Gulf Nuclear Site was designed for two units. Construction of second unit was halted prior to completion. Switch yard for both the units was completed

- SERI plans to use existing switchyard for the proposed unit(s)

- SERI has no plans to perform activities at ESP site under 10 CFR 50.10(e)(1), therefore no site redress plan is submitted

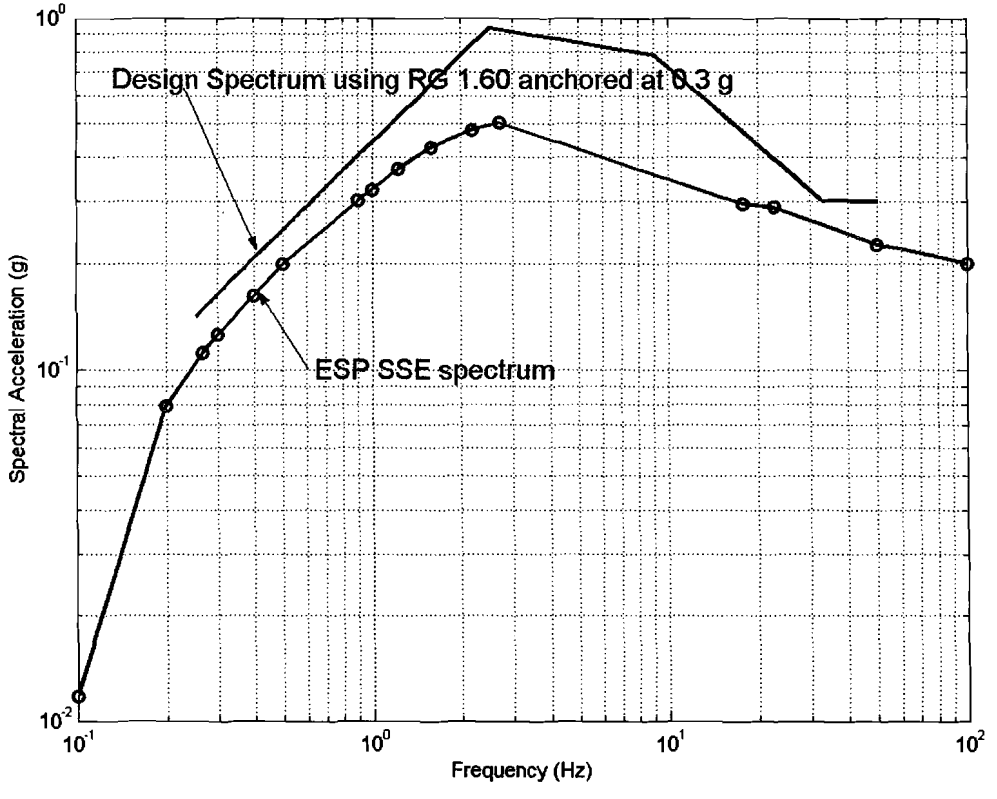

Proposed ESP
Exclusion Area Boundary
GGNS
Unit 1
Proposed ESP Facility
Footprint Area
2 miles
2760 ft
0.52 miles
Low Population
Zone (LPZ)
Property Boundary

# ESP Site Features

- Location of the ESP site
- Cooling water use
- Flooding in Mississipppi River
- Local Intense Precipitation
- Effects of  Probable Maximum Flood
- Flood Carrying Capacity of Mississippi River near the ESP site
- Effects of low water
- Ground water use

# Seismic

- The ESP is located in a relative low seismic region.
- No earthquake recorded within 25 miles radius since 1777.
- No active faults mapped within 90 miles radius
- The ESP site is a deep soil site
- Seismic hazard estimate using Regulatory Guide 1.165


Design Spectrum using RG 1.60 anchored at 0.3 g
ESP SSE spectrum
Spectral Acceleration (g)
Frequency (Hz)
$10^0$
$10^{-1}$
$10^{-2}$
$10^{-1}$
$10^0$
$10^1$
$10^2$

# Emergency Planning

- SERI has elected to seek acceptance of "major features" of emergency plans as provided in 10 CFR 52.17(b)(2)(i)
- NRC/FEMA have issued draft guidance document, Supplement 2 to NUREG-0654/FEMA-REP-1
- Generic industry concern with degree of finality associated with major features
- Staff can grant finality as to the overall description but will need to address implementation details at COL

# Future Milestones

- ACRS interim letter to the EDO assumed by end of June 2005
- Staff issues FSER on October 21, 2005
- Staff provides FSER to ACRS also in October 2005
- ACRS Subcommittee meeting for FSER on November 22, 2005, and Full committee meeting on December 8, 2005
- ACRS letter to the EDO assumed December 2005
- Staff incorporates ACRS letter and issues FSER as NUREG by 01/28/06
- Mandatory hearings begin early 2006
- Commission decision assumed October 2006

# Presentation Conclusions

- Staff issued DSER for SERI's ESP application on April 7, 2005
- Open item responses expected by June 21, 2005
- Looking forward to seeing interim ACRS letter and to briefing the Subcommittee and the full Committee on FSER during November/December, 2005

# Backup Slides

# Open Items

- 2.1-1, Control of exclusion area
  - Applicant must have control over exclusion area or irrevocable right to obtain control
  - Legal issue being addressed in Office of General Counsel

- 2.1-2, Weighted transient population data
  - The applicant needs to provide weighted transient population in the projected population density and population center

- 2.3-1, Applicant needs to provide 100-year return max/min dry-bulb temperatures

  values recorded at Jackson, MS during 1896-2003 (107°F/-5°F)
  - 110°F was recorded at Vicksburg MS (08/31/200)
  - -8°F was recorded at St. Joseph LA (01/27/1940)

# Open Items

- 2.3-2, Applicant needs to provide 48-hour probable maximum winter precipitation

- 2.3-3, Applicant needs to provide design basis site characteristic to assess potential for freezing in UHS water storage facility

- 2.3-4, Applicant needs to identify a 3-second gust wind speed that represents a 100-year return value used to determine wind loading

# Open Items

- 2.3-5, Applicant needs to identify Atmospheric Dispersion and Deposition Factors for nearest milk cow and meat cow

- 2.4-1, Applicant needs to provide corrected UTM coordinates of the center of the proposed powerblock and/or revise Figure 2.1-1 in the SSAR to show the correct location and coordinates.

- 2.4-2, Applicant needs to provide information on the elevation (depth) of the zone that could be disturbed by the construction of the new facility, such that the local subsurface environment and its alignment with the existing hydrogeological environment could be altered.

# Open Items

- 2.4-3, Applicant to provide more details regarding dewatering wells to allow the staff to determine whether ground surface subsidence could affect safety-related structures and piping. Provide information related to the location of dewatering wells in relation to safety-related structures and associated monitoring of the ground water table.
- 2.4-4, Applicant to provide more details regarding the floodwater level estimation, including data and methods used to arrive at the floodwater elevation of 133.25 feet MSL.

# Open Items

- 2.4-5, Applicant to revise and present estimates of the local intense precipitation as shown in Table 2.4-7 of the SSAR using the guidelines of HMR 52.
- 2.4-6, Applicant to provide further description of the rationale for considering Sr-90 and Cs-137 in the radionuclide transport analysis.
- 2.4-7, Applicant to factors, such as soil, sediment, and rock characteristics; adsorption and retention coefficients; ground water velocity; and distances to the nearest body of surface water are important to hydrological radionuclide transport. Provide these site characteristics from onsite measurements

# Open Items

- 2.5-1, Applicant to provide justification for not updating the background seismic source for the ESP site.
- 2.5-2, Applicant to provide and evaluate the criteria or weights used for ranking of model clusters and the judgements involved in balancing data consistency and adherence to seismological principles in the EPRI 2003 ground motion evaluation. Explain how recordings from a single earthquake can provide well-resolved values of both crustal quality factor (Q) and site kappa, also explain why the Q value of 317 at 1 Hz is much lower than values found in other studies of eastern North American earthquakes, and why other studies find less frequency dependence of Q in the eastern North American than in the western North American.

# Open Items

- 2.5-3, Applicant to provide an explanation why the magnitude and distance bin corresponding to the SRSZ makes no contribution to the hazard deaggregation
- 2.5-4, Applicant to provide justification on applying the generic shear wave velocity profile derived from Memphis area to the ESP site and on its applying kappa value derived from ground motion observation on the Mississippi embayment in the sensitivity test.
- 2.5-5, Applicant to provide the basis for the selection of values of BE, UB, and LB and other parameters for the base case profile.

# Open Items

- 13.3-1, Various additional details on offsite emergency response measures
  - Based on FEMA review of Supplement 2 acceptance criteria
- 13.3-2, Applicant responsibility for making information available to offsite authorities for distribution
- 13.3-4, Additional information on evacuation time estimate (ETE)
  - Clarify whether results of the 2003 ETE study were discussed with officials from the States of Mississippi and Louisiana involved in implementing traffic management plans

# Open Items

- 13.3-3, Adequacy of technical support center (TSC), emergency operations facility (EOF), and operations support center (OSC)
  - No evaluation or decision has been made by applicant as to whether the existing Unit 1 OSC and EOF facilities could or would be shared for proposed new reactor(s)
  - TSC facility would not be shared; Part 52 design certifications establish TSC design criteria, which would need to be incorporated as appropriate

Item #9, handout 1

# REGULATORY GUIDE FOR NFPA 805 RULE
## ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS
## JUNE 2, 2005

Sunil Weerakkody, Chief
Fire Protection Section
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

# OBJECTIVE AND OUTLINE

- Objective:
  - Seek ACRS endorsement to final regulatory guide on NFPA 805
- Outline:
  - Programmatic Overview – Paul Lain
  - Regulatory Guide – Robert Radlinski
  - Example of a Change Analysis using PRA and Fire Modeling – Ray Gallucci

2

# REGULATORY GUIDE FOR NFPA 805 RULE
## ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS
## JUNE 2, 2005

**Paul Lain, P.E.**
**Fire Protection Section**
**Office of Nuclear Reactor Regulation**
**U.S. Nuclear Regulatory Commission**
**Washington, DC 20555-0001**

# OBJECTIVE AND OUTLINE

- Objective: Seek ACRS endorsement of NFPA 805 Regulatory Guide
- Outline: Programmatic Overview
  - Regulatory Guide Schedule
  - Industry Interest

2

## Reg. Guide Schedule

| | |
|---|---|
| ■ Commission Approves Rule | June '04 |
| ■ ACRS Deferred DG Review | Sept. '04 |
| ■ DG Public Comment | Oct. '04 |
| ■ Public Meeting | Oct. '04 |
| ■ Public Meeting | Jan. '05 |
| ■ NEI 04-02, Rev. 0 | May '05 |
| ■ ACRS Subcommittee | May '05 |
| ■ CRGR Review | May '05 |
| ■ ACRS Full Committee | Jun. '05 |
| ■ Final Publication | July '05 |

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## Industry Interest

| | |
|---|---|
| ■ Duke Energy Letter of Intent (LOI) | Feb. '05 |
| ■ Oconee (Pilot) | 2005 to 2007 |
| ■ McGuire | 2006 to 2008 |
| ■ Catawba | 2007 to 2009 |
| ■ Progress Energy LOI (Tentative) | June '05 |
| ■ Harris (Pilot) | Complete '07 |
| ■ Brunswick | Complete '09 |
| ■ Crystal River 3 | Complete '10 |
| ■ Robinson | Complete '11 |

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# REGULATORY GUIDE FOR NFPA 805 RULE
## ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS
## JUNE 2, 2005

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**Fire Protection Section**
**Office of Nuclear Reactor Regulation**
**U.S. Nuclear Regulatory Commission**
**Washington, DC 20555-0001**

# SCOPE OF REGULATORY GUIDE

- Endorses NEI 04-02 and NEI 00-01 guidance documents
- Emphasizes key guidance issues
- Takes exception to Chapter 6 of NEI 04-02
- Identifies suggested fire models and provides guidance on fire PRAs
- Describes staff position on NFPA 805 appendices

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## PLANT CHANGES MODIFICATIONS AND DEVIATIONS

- Provides high-level guidance on screening of inconsequential changes and endorses NEI's more specific guidance for same
- Emphasizes need to perform integrated assessment of risk, defense-in-depth, and safety margin for all fire protection program changes
- Endorses NEI 04-02 guidance for use of deterministic approach, fire models, risk assessments or any combination of these

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## CIRCUIT ANALYSES

- Endorses NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis"
- Advocates addressing spurious actuations using risk-informed, performance-based methods
- Emphasizes that IN 92-18 type failures should be considered
- Provides guidance for addressing cumulative effects for multiple circuit analysis changes

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## OPERATOR MANUAL ACTIONS (RECOVERY ACTIONS)

- Explains that unapproved operator manual actions credited in III.G.2 areas should be transitioned as plant changes
  - Note that NFPA 805 requires that operator manual actions be evaluated using performance-based methods

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## Notable Change to NEI 04-02

- Only one significant change to NEI 04-02 since version provided for ACRS subcommittee meeting
- Provided additional guidance on what plant changes related to the fire protection program can be made without NRC approval

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# Conclusion

- The regulatory guide provides specific guidance on the implementation of an NFPA 805 fire protection program by endorsing the NEI guidance documents
- Provides appropriate clarification and emphasis for key issues
- Provides suitable guidance to licensees to assess impact of adopting a risk-informed, performance-based fire protection program

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Item #9, Handout 2

# PLANT CHANGE EVALUATION EXAMPLE FOR NFPA-805

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Presented to Full Committee of the ACRS, June 2, 2005



# PLANT CHANGE EVALUATION

- Licensee discovers two control cables for redundant HPI MOVs lying in same tray above electrical switchgear cabinets
  - Currently this CDF contribution has not been estimated since the case was not identified
  - Actual configuration contributes to fire CDF
  - Perform NEI-04-02 "Plant Change Evaluation" to determine acceptability under NFPA-805
    - $\Delta$CDF = CDF for unanalyzed fire scenario

# PRELIMINARY SCREEN

- Concern is that fire in switchgear cabinets could cause spurious closing of both HPI MOVs through damage to control cables
  - Perform "Preliminary Risk Screen" per Fig. 5-1 of NEI-04-02
    - Selected method = "Order-of-Magnitude" ΔCDF approximation via § 4.2 of NEI-00-01 ("Preliminary Screen for Risk Significance Analysis" in *Guidance for Post-Fire Safe Shutdown Circuit Analysis*)



# NEI-00-01 METHOD

- Preliminary Screen - NEI-00-01, § 4.2
  - Fire frequency (F)
  - Probability of spurious actuation (P)
  - Challenging fire factor (G)
  - Fire non-suppression probability (S)
  - CCDP (C)
  - Fraction for # of vulnerable fire zones (Z)
    - For simplicity, Z is ignored in this example
  - $\Delta CDF = F * P * G * S * C$

# FIRE RISK ESTIMATE (1)

- Fire frequency for Switchgear Room = MEDIUM (Table 4.5 in NEI-00-01)
- Prob. of spurious actuation = MEDIUM (assume T-plastic, inter-cable [Table 4.6])
  - F * P = MED * MED < 0.01/yr (Table 4-1)
- Challenging fire factor = 0.1 (large electrical fire/high-energy arc [Table 4-7])
  - F * P * G < 0.001/yr



# FIRE RISK ESTIMATE (2)

- Fire non-suppression probability = 1 (possibility of high-energy arcing fault, § 4.2.1.5)
  - F * P * G * S < 0.001/yr
- CCDP = 0.01 (0.1 given LOOP for internal events AND assume redundant shutdown equipment available for another 0.1 [Table 4-8])
  - F * P * G * S * C < 1E-5/yr
- "Does change (for bounding ΔCDF = 1E-5/yr) impact risk non-negligibly?" Yes → "Quantitative Risk Evaluation" in Fig 5.1 of NEI-04-02

# QUANTITATIVE EVALUATION

- Perform "sharpened-pencil" analysis using tools from Fire Protection (FP) SDP
- Component-based fire frequency
  - Assume cables pass horizontally above 10 switchgear cabinets → F = 10 * 6E-5/cabinet-yr = 6E-4/yr (Table A1.3 in FP SDP)
- Spurious actuation probability
  - Assume two T-plastic inter-cable interactions → $P = 0.2^2 = 0.04$ (Table 2.8.3)



# REFINED FIRE RISK EST. (1)

- Challenging fire factor
  - As before → G = 0.1 (large electrical fire/high-energy arcing fault at 650 kW [Table 2.3.1])
- Fire non-suppression probability
  - Consider physical layout and fire modeling
    - Horizontal cables within 5 ft of top of cabinet will be protected against high-energy arcing fault if tray is covered – add requirement to plant change (Att. 5)
    - Consider only effects of 650-kW electrical fire

## REFINED FIRE RISK EST. (2)

- Fire non-suppression probability (cont.)
  - Using correlations from NUREG-1805 (*Fire Dynamics Tools*) for physical layout and fire heat release rate, estimate temperature ≈ 500 ºF at cables → T-plastic cable failure in 10 min (Table A7.2)
    - Manual non-suppression probability for electrical fires → S = 0.3 for 10 min (Table A8.1)



## REFINED FIRE RISK EST. (3)

- CCDP
  - Take additional credit for Appendix R safe/ alternate shutdown strategies, including operator actions → C = 0.001 (Step 2.8)
- Recalculate ΔCDF
  - F * P * G * S * C = 6E-4/yr * 0.04 * 0.1 * 0.3 * 0.001 = 7E-10/yr
  - This satisfies ΔCDF < 1E-7/yr, but still must address defense-in-depth and safety margin
    - Parallel approach for ΔLERF

## D-IN-D/SAFETY MARGIN

- Uncertainty in fire ΔCDF can span several orders of magnitude, so even with best-estimate ΔCDF = 7E-10/yr, licensee must still consider defense-in-depth and safety margin
  - Magnitude of uncertainty may reflect degree of safety margin that is present and may suggest acceptable level of defense-in-depth



## CONCLUSION

- The Plant Change Evaluation concludes only after <u>all three</u> essential risk-related elements are satisfied, i.e., ΔCDF/ΔLERF, defense-in-depth and safety margin
  - Licensee must also consider cumulative effect of multiple plant changes, e.g., $\sum(\Delta CDF)_i$
  - Licensee documents Plant Change Evaluation according to plant procedures and maintains for review by NRC inspectors

Item #9, handout 3

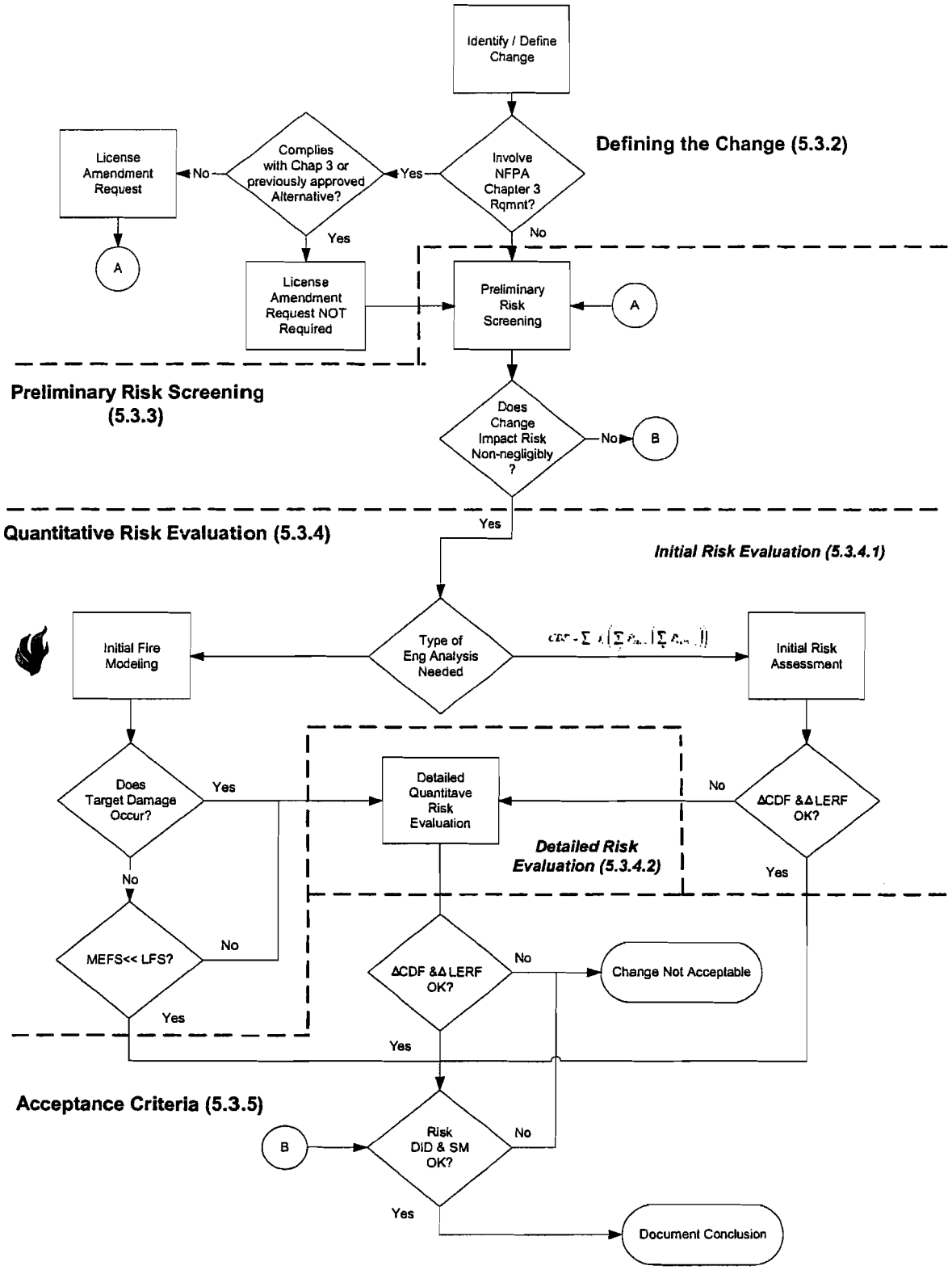


**Figure 5-1 - Plant Change Evaluation**