October 1, 2004

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: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF FEBRUARY 2002 OPERATIONAL CONDITIONS (TAC NO. MC4480)

Dear Mr. Bezilla:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) Program analysis of operational conditions that existed at Davis-Besse Nuclear Power Station from February 2001 until the plant was shutdown in February 2002 (Enclosure 1). The conditions involved the degraded vessel head, the cracking in the control rod drive mechanism nozzles, the unqualified coatings and debris in containment and the potential failure of high pressure injection pumps during recirculation as described in Licensee Event Reports 346/02-002, 346/02-005 and 346/03-002. The results of the preliminary ASP analysis indicate that this event is a significant precursor (i.e., conditional core damage probability $\geq 1 \times 10^{-6}$).

In assessing operational events, the Nuclear Regulatory Commission (NRC) staff strives to make the ASP models as realistic as possible regarding the specific features and response of a given plant to various accident sequence initiators. We realize that licensees may have additional systems and emergency procedures or other features at their plants that might affect the analysis. Therefore, we are providing you an opportunity to review and comment on the technical adequacy of the preliminary ASP analysis, including the depiction of plant equipment and equipment capabilities. Upon receipt and evaluation of your comments, we will revise the conditional core damage probability calculations where necessary to consider the specific information you have provided. The object of our review process is to provide as realistic an analysis of the significance of the event as possible. In order for us to incorporate your comments, perform any required re-analysis and prepare the final report of our analysis in a timely manner, you are requested to complete your review and to provide any comments within 60 calendar days from the date of this letter. As soon as our final analysis of this event has been completed, we will provide it and the resolution of your comments to you for your information.

We have also enclosed information to facilitate your review. Enclosure 2 contains specific guidance for performing the requested review, identifies the criteria which we will apply to determine whether any credit should be given in the analysis for the use of licensee-identified additional equipment or specific actions in recovering from the event and describes the specific information that you should provide to support such a claim.

This request is covered by the existing Office of Management and Budget clearance number (3150-0104) for NRC staff follow-up reviews of events documented in licensee event reports. Your response to this request is voluntary and does not constitute a licensing requirement.

The NRC staff is continuing to review the appropriate classification of these documents within our records management program considering changes in our practices following the events of September 11, 2001. Pending a final determination, the enclosed analyses have been marked as sensitive information. Therefore, the staff has not made it publicly available. Please control the document accordingly. We will inform you if the classification of the document changes as a result of our ongoing assessments. If you believe that your response to this letter includes potentially sensitive information, please discuss the matter with me prior to submitting the information.

Contact me if you have any questions.

Sincerely,

/RA/

Jon B. Hopkins, Senior Project Manager
Project Directorate III, Section 2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. Preliminary ASP Analysis (Sensitive - Not For Public Disclosure)
2. ASP Review Guidance

cc w/o enclosure 1: See next page

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DISTRIBUTION: w/o Enclosure 1 (Sensitive - NON-PUBLIC)

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OFFICIAL RECORD ONLY

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GUIDANCE FOR LICENSEE REVIEW OF PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an event or condition that occurred at your plant has been provided for your review. This analysis was performed as a part of the NRC's Accident Sequence Precursor (ASP) Program. The ASP Program uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage.

The types of events evaluated include actual initiating events, such as a loss of off-site power or loss-of-coolant accident, degradation of plant conditions, and safety equipment failures or unavailabilities that could increase the probability of core damage from postulated accident sequences.

This preliminary analysis was conducted using the information contained in the plant-specific final safety analysis report (FSAR), individual plant examination (IPE), and other pertinent reports, such as the licensee event report (LER) and/or NRC inspection reports.

Modeling Techniques

The models used for the analysis of events were developed by the Idaho National Engineering and Environmental Laboratory. The models were developed using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The developed models are called Standardized Plant Analysis Risk (SPAR) models. The SPAR models are based on linked fault trees. Fault trees were developed for each top event on the event trees to a super component level of detail.

Two revisions of the SPAR models are currently being used in the ASP analysis: SPAR Rev. 2 and SPAR Rev. 3.

- SPAR Rev. 2 models have four types of initiating events:

  - transients,
  - small loss-of-coolant accidents (LOCAs),
  - steam generator tube rupture (PWR only), and
  - loss of offsite power (LOSP).

The only support system modeled in Rev. 2 is the electric power system.

- SPAR Rev. 3 models are currently being developed to replace Rev. 2 models. The newer revision models have 11 types of initiating events:

  - transients,
  - small LOCAs,
  - medium LOCA,
  - large LOCA,
  - interfacing system LOCA,
  - steam generator tube rupture (PWR only),
  - LOSP,
  - loss of component cooling water (PWRs only),
  - loss of service water, and
  - loss of DC power.

Both revisions have transfer events trees for station blackout and anticipated transient without scram.

The models may be modified to include additional detail for the systems/components of interest for a particular event. This may include additional equipment or mitigation strategies as outlined in the FSAR or IPE. Probabilities are modified to reflect the particular circumstances of the event being analyzed.

Guidance for Peer Review

Comments regarding the analysis should address:

- Does the "Event Summary" section:
  - accurately describe the event as it occurred; and
  - provide accurate additional information concerning the configuration of the plant and the operation of and procedures associated with relevant systems?
- Does the "Modeling Assumptions" section:
  - accurately describe the modeling done for the event;
  - accurately describe the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions; and
  - include assumptions regarding the likelihood of equipment recovery?

Appendix G of Reference 1 provides examples of comments and responses for previous ASP analyses.

Criteria for Evaluating Comments

Modifications to the event analysis may be made based on the comments that you provide. Specific documentation will be required to consider modifications to the event analysis. References should be made to portions of the LER or other event documentation concerning the sequence of events. System and component capabilities should be supported by references to the FSAR, IPE, plant procedures, or analyses. Comments related to operator response times and capabilities should reference plant procedures, the FSAR, the IPE, or applicable operator response models. Assumptions used in determining failure probabilities should be clearly stated.

Criteria for Evaluating Additional Recovery Measures

Additional systems, equipment, or specific recovery actions may be considered for incorporation into the analysis. However, to assess the viability and effectiveness of the equipment and methods, the appropriate documentation must be included in your response. This includes:

- normal or emergency operating procedures,
- piping and instrumentation diagrams (P&IDs),
- electrical one-line diagrams,
- results of thermal-hydraulic analyses, and
- operator training (both procedures and simulation).

This documentation must be current at the time of the event occurrence. Systems, equipment, or specific recovery actions that were not in place at the time of the event will not be considered. Also, the documentation should address the impact (both positive and negative) of the use of the specific recovery measure on:

- the sequence of events,
- the timing of events,
- the probability of operator error in using the system or equipment, and
- other systems/processes already modeled in the analysis (including operator actions).

An Example of a Recovery Measure Evaluation

A pressurized-water reactor plant experiences a reactor trip. During the subsequent recovery, it is discovered that one train of the auxiliary feedwater (AFW) system is unavailable. Absent any further information regrading this event, the ASP Program would analyze it as a reactor trip with one train of AFW unavailable. The AFW modeling would be patterned after information gathered either from the plant FSAR or the IPE. However, if information is received about the use of an additional system (such as a standby

steam generator feedwater system) in recovering from this event, the transient would be modeled as a reactor trip with one train of AFW unavailable, but this unavailability would be mitigated by the use of the standby feedwater system.

The mitigation effect for the standby feedwater system would be credited in the analysis provided that the following material was available:

- standby feedwater system characteristics are documented in the FSAR or accounted for in the IPE,

- procedures for using the system during recovery existed at the time of the event,

- the plant operators had been trained in the use of the system prior to the event,

- a clear diagram of the system is available (either in the FSAR, IPE, or supplied by the licensee),

- previous analyses have indicated that there would be sufficient time available to implement the procedure successfully under the circumstances of the event under analysis, and

- the effects of using the standby feedwater system on the operation and recovery of systems or procedures that are already included in the event modeling. In this case, use of the standby feedwater system may reduce the likelihood of recovering failed AFW equipment or initiating feed-and-bleed due to time and personnel constraints.

Reference

1. R. J. Belles, et al., “Precursors to Potential Severe Core Damage Accidents: 1997, A Status Report,” USNRC Report NUREG/CR-4674 (ORNL/NOAC-232) Volume 26, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, and Science Applications International Corp., Oak Ridge, Tennessee, November 1998.