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Michael R. Kansler
President

May 17, 2004
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Indian Point Nuclear Generating Units Nos. 2 and 3
Docket Nos. 50-247, and 50-286
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Entergy Nuclear Operations, Inc.
Comments on Preliminary Accident Sequence
Precursor Analysis of August 14, 2003 Operational Event

References:

1. NRC letter, J. P. Boska to M. Kansler, dated March 18, 2004 regarding James A. FitzPatrick Nuclear Power Plant Re: Review of Preliminary Accident Sequence Precursor Analysis of August 14, 2003 Operational Event

2. NRC letter, J. P. Boska to M. Kansler, dated March 18, 2004 regarding Indian Point Unit 2 Nuclear Power Plant Re: Review of Preliminary Accident Sequence Precursor Analysis of August 14, 2003 Operational Event

3. NRC letter, J. P. Boska to M. Kansler, dated March 18, 2004 regarding Indian Point Unit 3 Nuclear Power Plant Re: Review of Preliminary Accident Sequence Precursor Analysis of August 14, 2003 Operational Event

Dear Sir:

Attached are Entergy Nuclear Operations', Inc. (ENO) comments on the NRC's preliminary Accident Sequence Precursor (ASP) Program analysis for the James A. FitzPatrick, Indian Point 2, and Indian Point 3 Nuclear Power Plants (Attachments 1, 2 and 3 respectively). The NRC staff asked ENO to review these analyses, which involve an August 14, 2003 operational event involving a reactor trip and loss of offsite power, and to provide written comments within 60 days (Reference 1, 2 and 3).

ENO agrees with the overall results of the analyses. However, the analyses do not consider all viable and effective accident recovery measures. In some cases, the analyses did not fully

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credit reasonably expected physical phenomena. Specific comments on selected accident sequences are detailed in the attachments.

There are no new commitments made in this letter. If you have any questions, please contact Ms. Charlene D. Faison at 914-272-3378.

Very truly yours,

Michael R. Kansler
President
Entergy Nuclear Operations, Inc.

cc: Next page

List of Attachments:

1. Comments on Preliminary Accident Sequence Precursor for FitzPatrick
2. Comments on Preliminary Accident Sequence Precursor for Indian Point 2
3. Comments on Preliminary Accident Sequence Precursor for Indian Point 3

cc:

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Comments on NRC Preliminary Accident Sequence Precursor for FitzPatrick

Introduction

Below are Entergy Nuclear Operations', Inc. (ENO) comments on the NRC's preliminary Accident Sequence Precursor (ASP) Program analysis for the James A. FitzPatrick nuclear power plant. The NRC staff asked ENO to review these analyses, which involve an August 14, 2003 operational event involving a reactor trip and loss of offsite power, and provide written comments within 60 days (Reference 1).

Comments

1. The NRC's Accident Sequence Precursor (ASP) analysis did not include the three-hour reactor coolant boil-off time, which would take place in a site blackout sequence after battery depletion with initially successful HPCI/RCIC. Crediting this time in the analysis would increase the maximum offsite power recovery time to 7 hours and reduce the Conditional Core Damage Probability (CCDP). The NRC analysis permitted a maximum recovery time of only 4 hours, i.e. 3 hours (when the 115kV system was first adequately re-energized), plus one hour to realign the safety buses, 10500 and 10600.

2. The crosstie between the firewater system and the EDG (Emergency Diesel Generator) jacket coolers can be credited in accident sequence cutsets which involve failures of components in the ESW (Emergency Service Water) system with successful closure of 46MOV-102A/B. As an example, for the dominant core damage sequence LOOP/SBO (Loss of Offsite Power/Station Black Out) sequence 47-02, fire water cross-tie recovery can be credited for cutsets of CCDP 6.5E-7 and below.

3. Although the above comments would tend to lower the total CCDP due to the LOOP event of August 14th., ENO determination of CCDP for this event, using the ENO PRA model for FitzPatrick, show a similar CCDP and precursor determination for these events.

Reference

1. NRC letter, J. P. Boska to M. Kansler, dated March 18, 2004 regarding James A. FitzPatrick Nuclear Power Plant Re: Review of Preliminary Accident Sequence Precursor Analysis of August 14, 2003 Operational Event

Comments on NRC Preliminary Accident Sequence Precursor for Indian Point 2

Introduction

Below are Entergy Nuclear Operations', Inc. (ENO) comments on the NRC's preliminary Accident Sequence Precursor (ASP) Program analysis for the Indian Point 2 (IP2) nuclear power plant. The NRC staff asked ENO to review these analyses, which involve an August 14, 2003 operational event involving a reactor trip and loss of offsite power, and provide written comments within 60 days (Reference 1).

Comments

1. The analysis includes cutsets that include equipment in maintenance. Moreover, many of the cutsets involve having more than one major component in maintenance simultaneously. The normal work planning process at IP2 would not schedule maintenance on these components during the same workweek.

   A more specific comment with respect to maintenance unavailability regards the inclusion of basic events representing service water pump maintenance. A significant number of the cutsets in the dominant sequence contain such events. The cooling of the emergency diesel generators (EDGs) is not unitized to the service water pumps. That is, failure (or maintenance) of a specific service water pump (in these cases SWS Pump 26) does not fail the EDG that powers it. Thus, for example, in the cutset in Table 3 that contains AFW-TDP-TM-22, EPS-DGN-FR-22 and SWS-MDP-TM-26, emergency diesel generator EDG 23 (and EDG 21) would continue to receive cooling water and therefore motor driven AFW Pump 23 (which is powered from EDG 23) will continue to be powered.

   In addition, test and maintenance activities are not normally done on service water pumps when they are aligned to the essential service water header. When pumps on the essential header require maintenance, the normal process is to re-align them to the non-essential header and then perform the maintenance. As a result, it is inappropriate to assign an average maintenance unavailability value to a cutset where the service water pump is intended to represent a pump aligned to the essential header. If any unavailability is assigned to service water pumps when they are aligned to the essential header it would only be for the brief period when a failure has occurred prior to realigning the headers. This would be at least an order of magnitude lower. (Service water system pump unavailability in the ASP is higher than the current plant specific unavailability for any of the service water pumps.)

2. In a number of the cutsets, it appears that the bleed and feed failure is a result of an emergency diesel generator that powers one of the block valves failing to run. Since the block valve will receive an open signal on rising primary system pressure almost immediately after the LOOP event, the mission time for the EDGs for those cutsets should be very short (no more than a few minutes). If such a mission time were applied, the frequency associated with those cutsets would be much lower.

3. Emergency diesel generator maintenance unavailability is high by a factor of two compared to recent plant-specific information.

4. The assumption that AC power must be recovered before battery depletion, in lieu of

Comments on NRC Preliminary Accident Sequence Precursor for Indian Point 2

continued operation of the turbine-driven AFW pump and no RCP seal LOCA, also seems overly conservative. While the restoration of offsite power without DC power is more difficult, it is not improbable. In addition, procedures exist for manually closing breakers in the event of a loss of DC power.

Reference

1. NRC letter, J. P. Boska to M. Kansler, dated March 18, 2004 regarding Indian Point 2 Nuclear Power Plant Re: Review of Preliminary Accident Sequence Precursor Analysis of August 14, 2003 Operational Event

Comments on NRC Preliminary Accident Sequence Precursor for Indian Point 3

Introduction

Below are Entergy Nuclear Operations', Inc. (ENO) comments on the NRC's preliminary Accident Sequence Precursor (ASP) Program analysis for the Indian Point 3 nuclear power plant. The NRC staff asked ENO to review these analyses, which involve an August 14, 2003 operational event involving a reactor trip and loss of offsite power, and provide written comments within 60 days (Reference 1).

Comments

1. It is not clear how Indian Point's Appendix R emergency diesel generator was modeled in the analysis. The dominant scenario in this analysis is a valid core damage scenario (i.e. failure of all emergency diesel generators (EDG) and subsequent failure to recover AC power). However, it was observed that AC power recovery takes credit for operator action to align the gas turbines, but no credit appears to be taken for use of the Appendix R diesel. The 2nd cutset in Table 3 for Sequence 19-02 appears to include successful operation of the Appendix R diesel, yet the cutset still results in failure. In fact, all the cutsets in which the Appendix R event appears involve success (/EPS-XHE-XM-APPENDR), not failure, of the Appendix R diesel. Furthermore, the 1st and 2nd cutsets for Sequence 19-02 have the exact same failures (OEP-XHE-XM-GTBD, OEP-XHE-NOREC-BD and EPS-DGN-CF-RUN), and only the successes are different. Therefore, the cutsets are not minimal.

2. If the Appendix R diesel is in fact modeled, it doesn't appear that any credit is taken for its success. It should be noted that in typical station blackout scenarios, the Appendix R diesel can be aligned to the normal 480V AC safeguards buses (i.e., 2A/3A, 5A or 6A) and not just the Appendix R safe shutdown bus (i.e., MCC 312A).

3. The assumption that AC power must be recovered before battery depletion, in lieu of continued operation of the turbine-driven auxiliary feedwater pump and no reactor coolant pump seal LOCA (loss of coolant accident), is overly conservative. Restoring offsite power without DC power is more difficult, it is not improbable. In addition, procedures exist for manually closing breakers in the event of a loss of DC power.

4. Some cutsets involve maintenance combinations that would not be permitted during plant operation. As an example, the 9th cutset in Sequence 19-02 (2.8E-7) includes maintenance unavailablity of 31 EDG simultaneously with maintenance of 36 service water pump (supplied by 32 EDG). The normal work planning process at IP3 would not schedule maintenance on these components simultaneously.

5. A more specific comment with respect to maintenance unavailability regards the inclusion of basic events representing service water (SW) pump maintenance. Test and maintenance activities are not normally done on SW pumps when they are aligned to the essential service water header. When pumps on the essential header require maintenance, the normal process is to re-align them to the non-essential header and then perform the maintenance. As a result, it is inappropriate to assign an average maintenance unavailability value to a cutset where the SW pump is intended to represent a pump aligned to the essential header. If any unavailability is assigned to SW pumps when they are aligned to the essential header it would only be for the brief period when a

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failure has occurred prior to realigning the headers. This would be at least an order of magnitude lower than the values used.

Reference

1. NRC letter, J. P. Boska to M. Kansler, dated March 18, 2004 regarding Indian Point 3 Nuclear Power Plant Re: Review of Preliminary Accident Sequence Precursor Analysis of August 14, 2003 Operational Event