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NOTE: Correct all copies of the applicable publication as specified below.

ITEM	REFERENCES (SECTION, PAGE PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
01	Page 5-39	Replace with new page 5-39
02	Page 5-39a	Replace with new page 5-39a
03	Page 5-39b	Insert new page 5-39b
04	Page 5-51a	Replace with new page 5-51a

5.5.2 Loss-of-Coolant Accident

This analysis of the Oyster Creek Nuclear Generating Station loss-of-coolant accident (LOCA) is provided to demonstrate conformance with the ECCS acceptance criteria of 10CFR50.46. The objective of the LOCA analysis contained herein is to provide assurance that the most limiting break size, break location and single failure combination has been considered for the plant. The documentation contained in this section is intended to satisfy these requirements.

The general description of the General Electric (GE) LOCA evaluation models is contained in Reference 5-18. Applicability and approval for pre-pressurized reload fuel are given in Reference 5-25. Model changes are described in References 5-20 and 5-21, which were approved by the USNRC (Reference 5-19). The analysis utilizes the short-term thermal-hydraulic model (LAMB) and the transient critical power model (SCAT) in addition to the long-term thermal-hydraulic model (SAFE) and the core heat-up model (CHASTE).

This LOCA analysis differs from previous BWR/2 LOCA analyses in the following ways:

1. Core flow coastdown and core depressurization are now calculated with the LAMB and SCAT codes, as opposed to no modelling of coastdown. In order to use the LAMB computer code for a non-jet pump plant, LAMB inputs were developed to execute the code for a recirculation line break. These inputs reflect the geometry of the non-jet pump plant. The LAMB code only allows for two recirculation pump loops; therefore, the five loops were modeled such that the intact loops are combined into one loop, and the broken loop is modeled as the second loop.

2. For the large break region, dryout time is now calculated using the SCAT code, as opposed to assuming a set dryout time.

3. Increased fission gas release at higher exposures has been included in the results.

4. As in the previous analyses, convective heat transfer coefficients specified in Appendix K are applied to the fuel rods and channel boxes

under spray cooling when the reactor pressure blows down to 125 psia, at which time rated core spray system flow is achieved. These convective coefficients were previously substantiated by demonstrating that there is sufficient bundle spray flow to justify the coefficients. This was based on full sparger spray distribution tests in air. Results of these tests were considered applicable to core spray performance in a reactor steam environment since the steam effect on spray distribution was believed to be small. Recently, a study using currently approved core spray design methodology (Reference 5-38) was performed to calculate in detail the spray distribution performance of ring spargers in steam (Reference 5-39). The results of this study indicate that adequate spray distribution is provided for reactor pressures up to 55 psia. For higher reactor pressures experimental test data are not available to extrapolate the spray distribution methodology to higher pressures in steam. Therefore, there are some uncertainties in the applicability of the spray distribution methodology for reactor pressures greater than 55 psia. For large break LOCAs, the reactor blows down below 55 psia before credit for core spray heat transfer is assumed. Therefore, uncertainties in spray distribution predictions in high pressure steam have no impact on these results. For some small breaks, however, the reactor blows down more slowly and spray heat transfer credit is assumed above 55 psia. For these small break conditions, LOCA sensitivity studies were performed to assess core cooling effects in this pressure range. These studies demonstrated that the inherent core cooling predicted during the blow down from 125 to 55 psia due to steam cooling, more than justifies the application of the Appendix K convective heat transfer coefficients. Additionally, there is a further core cooling effect due to core reflooding calculated for these small breaks. Since heat transfer from steam cooling or reflooding is conservatively neglected for Appendix K applications, the actual heat transfer coefficients still exceed those allows by 10CFR50 Appendix K. Therefore, analysis via approved methodology using Appendix K heat transfer coefficients remains a valid licensing basis for Oyster Creek.

The Oyster Creek LOCA analysis reported here was performed as an independent self-contained analysis similar to that performed as a lead plant analysis.

5.5.2.1 Input to Analysis

A list of the significant input parameters to the LOCA analysis is presented in Table 5-10.

5.5.2.2 LAMB ANALYSIS

This code is used to analyze the short-term blowdown phenomena for large postulated pipe breaks (breaks in which nucleate boiling is lost before the water level drops and uncovers the active fuel). The LAMB output (core flow as a function of time) is input to the SCAT code for calculation of blowdown heat transfer.

The LAMB results presented are:

- Core Average Inlet Flow Rate (normalized to unity at the beginning of the accident) following a Large Break.

5.5.2.3 SCAT ANALYSIS

This code completes the transient short-term thermal-hydraulic calculation for large breaks. The GEXL correlation is used to track the boiling transition in time and location. The post-critical heat flux heat transfer correlations are built into SCAT, which calculates heat transfer coefficients for input to the core heatup code (CHASTE).

The SCAT results presented are:

- Minimum Critical Power Ratio following a Large Break.

- Convective Heat Transfer Coefficient following a Large Break.

5.5.2.4 SAFE ANALYSIS

This code is used primarily to track the vessel inventory and to model ECCS performance during the LOCA. The application of SAFE is identical for all

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5-30 Letter, J. F. Quirk (GE) to T. P. Speis (NRC), "ODYN Improvements", October 13, 1981.

5-31 Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "Change in GE Methods for Analysis of Cold Water Injection Transients", September 30, 1980.

5-32 Letter, R. C. Tedesco (NRC) to G. G. Sherwood (GE), "Acceptance for Referencing General Electric Licensing Topical Report NEDO-24154/ NEDE-24154P", February 4, 1981.

5-33 Letter, R. H. Buchholz (GE) to P. S. Check (NRC), "ODYN Adjustment Methods for Determination of Operating Limits", January 19, 1981.

5-34 Letter, R. E. Engel (GE) to T. A. Ippolito (NRC), "End of Cycle Coastdown Analyzed With ODYN/TASC", September 1, 1981.

5-35 Letter, H. C. Pfefferlen (GE) to D. G. Eisenhut (NRC), "Correction of ODYN Errors", June 8, 1982.

5-36 *General Electric Company Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K, Amendment No. 2 - One Recirculation Loop Out-of-Service,* General Electric Company, Revision 1, July 1978 (NEDO-20566-2).

5-37 Letter, R. E. Engel (GE) to D. B. Vassallo (NRC), "Control Rod Drop Accident," February 24, 1982.

5-38 S. A. Sandoz, et al., "Core Spray Design Methodology Confirmation Tests," General Electric Co., March 1983 (NEDO-24712-A).

5-39 "Performance Evaluation of the Oyster Creek Core Spray Sparger," General Electric Co., January 1984 (Class III), (NEDE-30010).

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