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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
INTERIM RELIABILITY EVALUATION PROGRAM (IREP) REPORT
BALTIMORE GAS AND ELECTRIC COMPANY
CALVERT CLIFFS UNIT NO. 1
DOCKET NO. 50-317

Introduction

The report entitled "Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," NUREG/CR-3511, March 1984, is an evaluation of Calvert Cliffs Unit 1 using probabilistic risk assessment (PRA) techniques. The IREP was undertaken by Sandia National Laboratory for the Nuclear Regulatory Commission in order to further the application of PRA techniques to nuclear reactor safety analyses. Although the IREP was not initiated as a result of any particular safety concern, the results of the study did yield several interesting insights into the Calvert Cliffs facility, including the identification of significant single failures.

Discussion and Evaluation

The Calvert Cliffs IREP PRA analysis used fault tree and event tree models as the primary tools to evaluate the risk due to a core melt at Calvert Cliffs Unit 1. Core melt sequences initiated by one of three break-size loss-of-coolant accidents (LOCAs) or one of six categories of transients were evaluated, and the dominant (i.e., highest frequency) sequences were further analyzed to estimate the magnitude and frequency of radionuclide release.

The most significant sequences contributing to the core melt frequency are (1) Anticipated Transients Without Scram (ATWS) (33% of the total core melt frequency), (2) Small-small LOCAs (i.e., 1.9" to 3" in diameter) with makeup system failure in the recirculation phase (20% of the total core melt frequency), (3) the loss of a DC bus followed by failure of secondary heat removal (16% of the total core melt frequency), and (4) Loss of Offsite Power followed by failure of secondary heat removal or a stuck open relief valve (12% of the total core melt frequency). Transients involving (1) loss of power conversions and secondary steam release systems (PCS) (6% of the total core melt frequency), (2) transients requiring primary system pressure relief not including ATWS (1% of the total core melt frequency) and (3) all other transients (5% of the core melt frequency), were judged to be of lesser significance.

By letter dated January 22, 1985, the staff requested that BG&E review the IREP report and provide comments. The response by BG&E, in their letter dated March 26, 1985, indicated that they had reviewed NUREG/CR-3511 and had made certain system/procedural changes.

° In the sequence involving loss of PCS followed by loss of auxiliary feedwater (AFW), the mispositioning of valve AFW-161 was the most significant contributor. This valve is located at the condensate storage tank discharge.

The licensee indicated that this valve at both units (1-AFW-161 and 2-AFW-161) had been modified by removing the valve internals, thus eliminating this potential failure. In addition, the licensee indicated that procedures to use the AFW cross connect valve (MOV-4550) had been incorporated into Abnormal Operating Procedure 3D, Section III, thus decreasing the likelihood of total loss of AFW.

° For large break LOCAs, recirculation of coolant from the containment sump to the reactor coolant system via the high pressure safety injection (HPSI) pumps is credited. In the event of HPSI failure, the low pressure safety injection (LPSI) pumps may be used if reactor coolant system pressure is sufficiently lowered. Use of LPSI would require reset of the recirculation actuation signal (RAS) since RAS trips the LPSI pumps. The licensee has installed a key locked switch to override the RAS signal to the selected LPSI pump to facilitate post-RAS use of LPSI pumps.

° For LOCAs, the salt water system (SWS) provides the ultimate heat sink (Chesapeake Bay.) Equipment which must function post-LOCA and which requires cooling is serviced by one of two systems (Service Water System (SRWS) or Component Cooling Water System (CCWS) which in turn rejects heat to the SWS via two CCWS or two SRWS heat exchangers.) The ECCS Room coolers reject heat directly to the SWS. The common discharge on the salt water side of both CCWS heat exchangers passes through valve SWS-196; failure of SWS-196 in the closed position would result in loss of containment spray, HPSI, and LPSI. Similarly, the common discharge on the salt water side of the SRWS heat exchanger passes through SWS-197; failure of SWS-197 in the closed position would result in loss of diesel generators (during a LOCA with loss of offsite power) and the containment fan coolers. Accordingly, valves SWS-196 and SWS-197 at Calvert Cliffs Units 1 and 2 have been modified by removal of their internals to prevent failure in the closed position.

° For the station blackout event, it is assumed that both off-site and onsite power are unavailable. Core melt would result from failure of secondary heat removal. Calvert Cliffs has an available NRC-approved, alternate source of off-site power which consists of a 69KV line from the Southern Maryland Electrical Cooperative (SMECO). The IREP report states that, "In light of the importance of loss of offsite power sequences, in general, and station blackout in particular, the utility is reviewing its procedure for connecting the 69KV line. Also, as a result of Task Action Plan A-44 Station Blackout, it is likely that all plants will be required to have improved loss of offsite power procedures. These improved procedures and other changes should result in significant mitigation of loss of offsite power sequences in the future." Procedures for use of the 69KV SMECO line have been developed by BG&E and incorporated in Operating Instruction 27E, Section IV.

Although some of the system/procedural changes described in BG&E's March 26, 1985 letter were developed as a result of other programs, these changes are consistent with the IREP findings and result in improvements to the safety of the Calvert Cliffs facility.

Conclusion

There was nothing found in the Calvert Cliffs PRA results to indicate a need for NRC action. There are no apparent "outlier" risk dominant sequences in the PRA that suggest such a need. The core melt frequency for Calvert Cliffs Unit 1 was, in fact, similar to values predicted by PRA for other pressurized water reactors. Improvements made to Calvert Cliffs, including those described herein, decrease the core melt frequency still further.

With regard to the usefulness of the IREP results, any assessment of facility safety should be made in light of the overall limitations of the PRA technique; thus, use of PRA results out of PRA context is clearly inappropriate.

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