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Probabilistic Analysis of Allowed Outage Times Relaxation at a PWR Plant

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Technical Specifications (TS) in a nuclear power plant are specific requirements on its day-to-day operation, designed to protect public health and safety. Two primary aspects of the TS are (1) limiting conditions of operation (LCO) with allowed outage times (AOTs) and (2) surveillance testing intervals (STIs). In recent years, there has been growing interest[1-3] in the nuclear community in reexamining the TS. One of the reasons is that a significant portion of reactor downtime (plant unavailability) is attributable to the strict TS. Existing TS were derived from engineering judgement based on deterministic review; they were not directly risk-based, and their efficacy in enhancing public safety is difficult to establish.

This paper presents a summary of a critical review[4] of the Westinghouse report[5] which proposed that AOTs for a number of safety systems at the Byron Generating Station be increased from 3 to 7 days.

Basic methods that address the AOT problem in Ref. 5 are the usual PRA techniques that can be characterized as the methods of "static" fault trees. The static fault tree method is a technique used to evaluate higher-level performance measures of a plant, e.g., core damage frequency and health risks, by propagating, through a set of binary structure functions, time-averaged (over some baseline periods) performance measures of lower-level basic components. It is simple to use and generally conservative in comparison with other methods that can be used for the AOT problem.[6]

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Event trees used in Ref. 5 are slightly modified and expanded versions of those for the Zion Probabilistic Safety Study,[7] and detailed fault trees were developed for the Byron-specific safety systems. Ref. 5 employed a simplified support state approach, did not explicitly treat loss of service water as an initiating event, and only implicitly included the dependence of risk on AOT by using mean time to repair (MTTR).

Important methodological differences in the present work are: (1) it adopted a fault tree linking approach to provide minimal cutsets, in order to be sure that support systems were properly treated and that system failures were properly conditioned on initiating events; (2) it included the loss of service water initiator; and (3) it computed risk as functions of repair outage times (T_r's) between two bounding values. It is noteworthy that the fault tree linking approach provides holistic information for the top event and an integrated model which facilitates sensitivity studies, e.g., through importance analysis.

The evaluation in this study focused on the core damage frequency from one unit operation only, since crossties of the shared systems between the two units, in particular, the Essential Service Water System (ESWS) and the Component Cooling Water System (CCWS), were not completely established. Note that the ESWS at Byron Unit 1 consists of two trains with a single pump in each train. Figure 1 shows the results on core damage frequency as a function of T_r between AOT1 (19 hours) and AOT2 (168 hours) and depending on which system is relaxed in its AOT. The ESWS has a predominant effect. This is attributed to the loss of service water initiator included in this study where the probability of a reactor coolant pump (RCP) seal LOCA given loss of service water was nominally assumed to be 0.5. (See Ref. 4 for the results when it is assumed that there is no RCP seal LOCA given loss of service water.)

Figure 1 also shows the individual system rankings with regard to the AOTs: the ESWS and the diesel generators (DGs) are the first two dominant contributors to the increments in core damage frequency due to the AOT relaxation and next in importance is the Auxiliary Feedwater System (AFWS). The effects on the core damage frequency of the AOTs for the containment heat removal systems

(Containment Spray System and Containment Fan Coolers) and the ECCSs (Charging pumps, Safety Injection pumps, and RHR pumps) are considered to be small.

This study identified a significant vulnerability in the ESWS at Byron Unit 1 (and subsequently led to interim resolution, pending completion of Unit 2 for crosstie, that at least one ESW pump train from Unit 2 be made available for Unit 1) and supported NRC recommendation that AOTs for six systems (Containment Fan Coolers, Containment Spray System, Charging pumps, Safety Injection pumps, RHR pumps, CCWS) of insignificant importance be relaxed, while AOTs for three systems (ESWS, DGs, AFWS) not be relaxed.

References

1. Boccio, J. L., Fragola, J. R., Hall, R. E., Lofgren, E. V., Samanta, P. K., and Vesely, W. E., "Program Plan for a Procedure for Evaluating Technical Specifications (PETS)," Brookhaven National Laboratory, October 1984.

2. Vesely, W. E., Gaertner, J. P., and Wagner, D. P., "Methodology for Risk-Based Analysis of Technical Specifications," Paper No. 32, Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Safety Methods and Applications, Volume 1, February 1985, San Francisco, CA.

3. Papazoglou, I. A. and Cho, N. Z., "A Markovian Analysis of Limiting Conditions of Operation for the Reactor Protection System," Paper No. 36, Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Safety Methods and Applications, Volume 1, February 1985, San Francisco, CA.

4. Cho, N. Z., Chu, T-L., Xue, D., Bozoki, G., and Youngblood, R. W., "Analysis of Allowed Outage Times at the Byron Generating Station," Brookhaven National Laboratory, NUREG/CR-4404, BNL-NUREG-51930, April 1986.

5. Butler, J. C. et al., "Byron Generating Station Limiting Conditions for Operation Relaxation Program," Westinghouse Electric Corporation, WCAP-10526, Volumes 1 and 2, April 1984.

6. Cho, N. Z., Bozoki, G. E., and Youngblood, R. W., "Comparison of Methods Applicable to Evaluation of Nuclear Power Plant Technical Specifications,"

Proceedings of the International ANS/ENS Topical Meeting on Thermal Reactor Safety, Volume 2, February 1986, San Diego, CA.

7. Commonwealth Edison Company, "Zion Probabilistic Safety Study," NRC Docket Nos. 50-295 and 50-304.

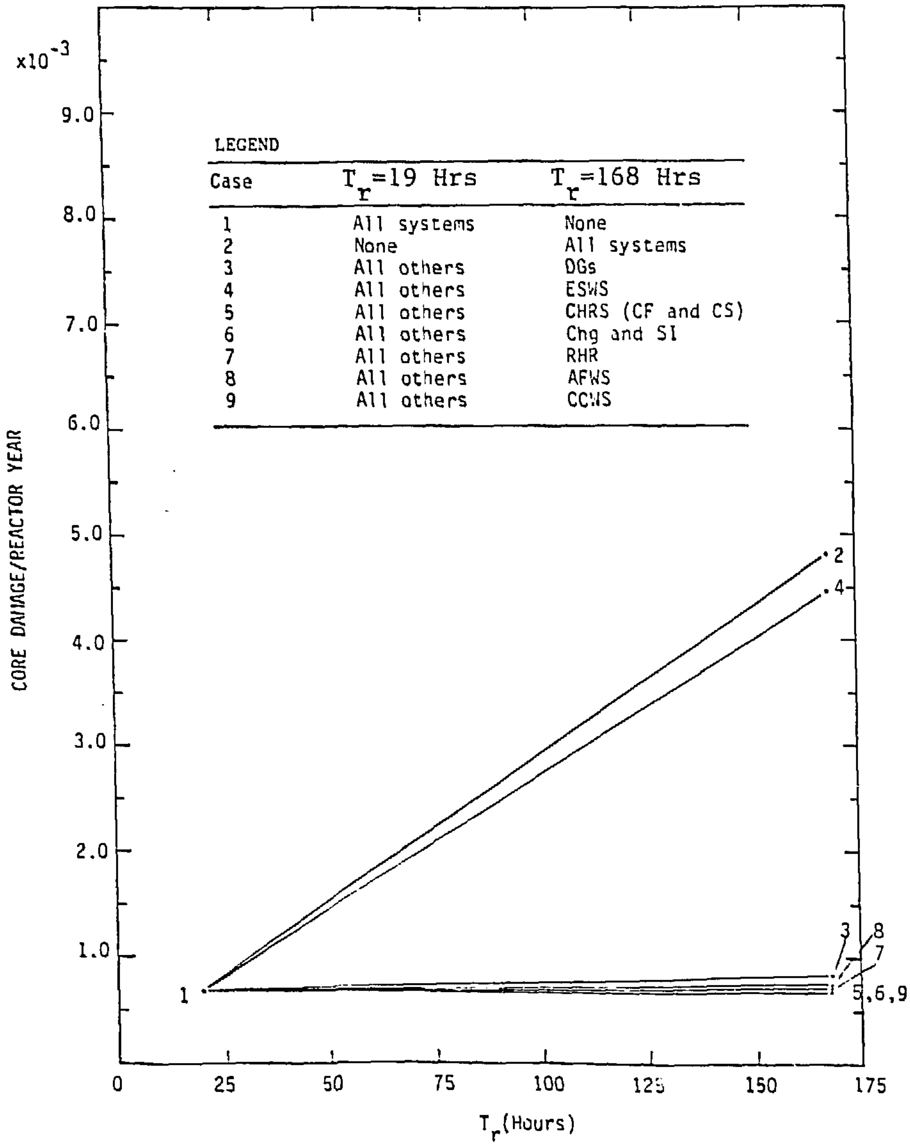

Figure 1. Core damage frequency as a function of T_r's for 9 cases assuming the probability of a RCP seal LOCA given loss of service water is 0.5.