APPENDICES

APPENDIX A

TECHNICAL BASIS OF THE STATION BLACKOUT RULE AND ADDITIONAL BACKGROUND

APPENDIX A

Technical Bases of the Station Blackout Rule and Additional Background

Technical Bases of the Station Blackout Rule Rule

The SBO rule evolved from the results of several plant-specific probabilistic safety studies; operating experience; and reliability, accident sequence, and consequence analyses completed between 1975 and 1988. WASH-1400, "Reactor Safety Study," 1975, indicated that SBO could be an important contributor to the total risk from nuclear power plant accidents. This study concluded that if an SBO persists for a time beyond the capability of the ac-independent systems to remove decay heat, core melt and containment failure could follow. In 1980, the Commission designated the issue of SBO as USI A-44, "Station Blackout," and the staff completed several technical studies to determine if any additional safety requirements were needed.

"Evaluation of Station Blackout at Nuclear Power Plants," NUREG-1032, June 1988 [Ref. 1], integrated the findings of the technical studies completed for USI A-44. NUREG -1032 presented the staff's major technical findings for the resolution of USI A-44, and provided the basis for the SBO rule and the accompanying RG 1.155, "Station Blackout," August 1988 [Ref. 2].

The NUREG-1032 evaluation of EDG train reliability used results and data from NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear Power Plants," July 1983 [Ref. 3]. NUREG/CR 2989 used the fault trees from 18 site PRAs/IPEs to identify the EDG failure boundary and classify failures. The boundary of the EDG train for the purposes of analyzing failures included a single engine, a generator, an output circuit breaker, and support subsystems necessary to power and sequence electrical loads to the vital bus. Consistent with the licensee PRAs/IPEs, the NUREG-1032 analyses of EDG unreliability considered planned and unplanned EDG demands and failures to start and load-run, EDG unavailability due to test and MOOS while the reactor was in power and non-power status, EDG failure recovery, and EDG common-cause failures. PRA/IPE reliability fault trees models included an unavailability contribution from MOOS since the EDG cannot perform its safety function upon demand if it is out of service. EDG MOOS while the reactor is at power can be an important consideration since the plant risk is potentially higher due to the possibility of a demand while the EDG is unavailable. EDG unavailability measurement can be based on the hours the EDG is unavailable or on the number of failures per demand. Both measures are unbiased estimates of EDG unavailability and are comparable so long as both measures are based on the same considerations (i.e., both consider MOOS).

The EDG analyses characterized the safety function of the EDGs as ability to start and load on demand, and used actual test and unplanned demands and failures (in excess of 10,000 demands and 100 failures) to measure reliability. NUREG-1032 found EDG reliability to be 0.98 or better and MOOS to be an average of 0.006 with range of 0–0.037. The NUREG-1032 data indicates the number of EDG failures on demand with an EDG in MOOS was 0.0056, which is approximately the same as 0.006 based on the time the EDGs were unavailable.

In March 1986, the NRC issued draft RG 1.155, which presented an acceptable method to comply with the SBO rule based on plant-specific characteristics and the dominant risk factors from NUREG-1032. The RG 1.155 was issued August 1988 and provided for selection of the SBO coping duration based on plant-specific characteristics including past unit average EDG train performance criteria and emergency ac power system configuration. The past unit average EDG train performance criteria were based on a reliability of greater than 0.90, 0.94, and 0.95 in the last 20, 50, and 100 demands. In general, the plants could select the 0.975 EDG target reliability level to achieve shorter coping durations. Plants with the lowest level of independence and redundancy in onsite emergency power supplies for safe shutdown equipment had to select the 0.975 minimum EDG target reliability level.

RG 1.155 contains guidance on (1) maintaining an individual EDG target reliability of 0.95 per demand or 0.975 per demand and assumes that as long as the MOOS unavailability is not excessive, the maximum EDG failure rate would result in overall reliability for the emergency power system), (2) establishing an EDG reliability program with test, maintenance, data collection, and management oversight elements to maintain the selected EDG target reliability level, (3) developing procedures and training to cope with an SBO, (4) selecting a plant-specific minimum acceptable SBO duration capability of either 2, 4, 8, or 16 hours based on plant-specific considerations, (5) evaluating SBO capability based on the selected duration capability, and (6) completing modifications to cope with an SBO. RG 1.155 expected that the individual EDG reliability would be calculated per NSAC-108, "The Reliability of Emergency Diesel at U.S. Nuclear Power Plants," September 1986 [Ref. 4], as the product of start reliability and the load-run reliability and did not consider MOOS. The calculation would use data that met the NSAC-108 definition for valid EDG start and load-runs.

Public and Industry Comment

In March 1986, a Notice of Proposed Rulemaking on SBO was published in the *Federal Register* (51 FR 9829), "Station Blackout." That notice invited public comments on a proposed SBO rule, draft RG 1.155, and NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, 'Station Blackout'," June 1988 [Ref. 5]. Among the 53 letters that were received commenting on the SBO rule was a critique of the backfit analysis that pointed out numerous errors, omissions, and inaccuracies. In addition, an industry comment stated that the costs would be much higher than those calculated by the NRC. The NRC addressed the cost concerns in a report, "Response To Industry Comments on Station Blackout Cost Analysis (NUREG/CR-3840)," November 1986 [Ref. 6], by increasing most of the cost items by 20 to 140 percent. Consequently the NUREG-1109 industry costs are those provided by the industry. The other comments were addressed in the final proposed resolution to USI A-44 that was reviewed and approved by the Committee to Review Generic Requirements in May 1987 and ACRS in June 1987.

In November 1987, NUMARC (subsequently renamed NEI) submitted NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," November 1987 [Ref. 7], as an alternative to comply with the SBO rule. NUMARC 87-00 content followed RG 1.155 and added very prescriptive, practical guidance to help the industry implement RG 1.155. NUMARC 87-00 addressed risk reduction by requiring the following: (1) taking action to reduce the risk if the licensees fell into the 8- or 16-hour coping category; (2) establishing procedures to cope with an SBO, restore ac power following an SBO, and to prepare for severe weather; (3) reducing EDG cold fast starts for testing; and

(4) monitoring AC power unavailability by providing data to the Institute of Nuclear Power Operations on a regular basis. By reference in RG 1.155, the staff concluded that NUMARC 87-00 (dated November 1987) contains guidance acceptable to the staff for meeting the SBO rule. Since 1989, the industry has used the EPS unavailability as an industry safety system performance indicator.

The staff issued SECY-88-22, “Final Station Blackout Rule, USI A-44," January 21, 1988 [Ref. 8], to obtain Commission approval for publishing the Notice of Final Rulemaking on the subject of SBO. In SECY 88-22, the staff recommended that the Commission issue the SBO rule, NUREG-1032, RG 1.155, and NUREG-1109 which documented the evaluation of five alternatives to close USI A-44 and the value-impact analysis of the proposed SBO rule. In SECY-88-22, the staff stated that USI A-44 was related to such other GSIs as GSI 56, “Diesel Generator Reliability”; USI A-45, “Shutdown Decay Heat Removal Requirements”; GSI 23, “Reactor Coolant Pump Seal Failures”; and GSI 128, “Electric Power Reliability.” In SECY-88-22, the staff states that any additional requirements or guidance contained in the resolutions of these GSIs must be consistent with the requirements of the SBO rule and were not expected to cause licensees to revise analyses, procedures, or equipment that were changed to comply with the SBO rule.

Resolution of Generic Safety Issue 56

The NRC SBO initiatives planned that the resolution of GSI 56 would detail an EDG reliability program consistent with RG 1.155. SECY-93-044, “Resolution of Generic Safety Issue B-56, Diesel Generator Reliability,” February 22, 1993 [Ref. 9], discussed several options and implementation of the recommendations as follows:

(1) Incorporate (by reference or example) EDG maintenance unavailability operating experience and the NUMARC EDG reliability program (Appendix D of NUMARC 87-00, Rev. 1, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,” August 1991 [Ref. 10]), into the regulatory guide and NUMARC guideline being developed for the maintenance rule, “NUMARC 93-01 Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” May 1993 [Ref. 11]. Appendix D of NUMARC 87-00, Rev. 1 (Section D.2.3.), employs trigger values to indicate when EDGs do not meet the selected target reliabilities. This recommendation was implemented by RG 1.160, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” June 1993 [Ref. 12]. However, Note 3 of RG 1.160 indicated that the triggers are intended to indicate when EDG performance problems exist such that additional monitoring is necessary and that it is not practical to demonstrate by statistical analysis that conformance to the trigger values will ensure the attainment of high reliability with reasonable confidence, of individual EDG units. By 1993 many licensees adopted the NUMARC trigger values as a means to meet the 0.95 and 0.975 EDG target reliability.

The maintenance rule (10 CFR 50.65) and the accompany regulatory guide RG 1.160 provide for the maintenance of safety-related structures, systems, and components (SSCs). Paragraph (a)(1) of 10 CFR 50.65 requires that power reactor licensees monitor the performance or condition of SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. Paragraph (a)(2) of 10 CFR 50.65 states that

monitoring as specified in paragraph (a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through preventive maintenance, so that the SSC remains capable of performing its intended function. Paragraph (a)(3) of 10 CFR 50.65 requires that performance and condition monitoring activities and associated goals and preventive maintenance activities be evaluated at least every refueling cycle provided the interval between evaluations does not exceed 24 months. RG 1.160 is the regulatory guide that accompanied the maintenance rule which is used in part to evaluate the effectiveness of EDG maintenance activities associated with compliance with the SBO rule.

(2) Issue a generic letter to allow licensees to voluntarily adopt the accelerated testing provisions in the improved standard technical specifications. The generic letter would further explain that, upon determination that the maintenance program conforms to the applicable approved guidance for diesel testing, the accelerated testing requirements in the technical specifications may be relocated to the maintenance program. This recommendation was implemented by the issue of GL-94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," May 31, 1994, which allowed licensees to eliminate accelerated testing requirements provided the licensee committed to early implementation of the maintenance rule consistent with the guidance of RG 1.160 with the exception of the reference to NUMARC 87-00, Rev, 1, trigger values. To date approximately 16 of the 64 nuclear power plants with accelerated test requirements have licensing amendments to delete accelerated testing requirements; however, the amendments were dated after the implementation of the maintenance rule, so it appears there was no need to consider the trigger values exception.

(3) Revise RG 1.9, Rev. 2, to incorporate elements of RG 1.108, Rev. 2 and GL 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," dated July 1984. This recommendation was implemented with the issue of RG 1.9, Rev. 3 , dated July 1993.

Regulatory Follow-up

The SBO rule requires that the NRC staff complete a regulatory assessment and notify the licensees of the staff's conclusions regarding the licensees' response to the SBO rule. The NRC completed safety evaluations for each plant that can be found in the NRC Public Document Room under each plant's docket number.

To assess the industry's compliance in implementing the SBO rule, the NRC completed eight pilot inspections by October 1994 using Temporary Instruction 2515/120, "Inspection of Implementation of Station Blackout Rule," (no date) [Ref. 13]. The objective of Temporary Instruction 2515/120 was to verify the adequacy of licensee programs, procedures, training, equipment and systems, and supporting documentation for implementing the SBO rule. The inspectors found that, overall, the licensees for the eight sites had satisfactorily implemented their commitments for conforming to the SBO rule. The inspections uncovered minor weaknesses in plant SBO documentation, primarily in the areas of electrical calculations and procedures that were considered to have no effect on SBO mitigation and to be insignificant. On the basis of these inspections, the staff concluded that it need not conduct additional team inspections to verify licensee implementation of the SBO rule. However, additional

discretionary SBO inspections could be conducted to verify that the licensees have taken appropriate actions to comply with the SBO rule.

Inspection Procedure 62706, “Maintenance Rule,” December 31, 1997 (Section 3.05, “Effectiveness of Emergency Diesel Generator (EDG) Maintenance Activities”) [Ref. 14], provides additional specific guidance that NRC inspectors should verify.

SBO rule assessments have been completed under other NRC programs and these were used in this report. For example, the NRC staff evaluated the impact of the SBO rule on CDFs in NUREG-1560. In SECY 97-180 the staff also evaluated the industry’s average cost per person-rem averted in satisfying the SBO requirements.

REFERENCES

1. U.S. Nuclear Regulatory Commission, “Evaluation of Station Blackout at Nuclear Power Plants,” NUREG-1032, June 1988.

2. U.S. Nuclear Regulatory Commission, “Station Blackout,” Regulatory Guide 1.155, August 1988.

3. U.S. Nuclear Regulatory Commission, “Reliability of Emergency AC Power Systems at Nuclear Power Plants,” NUREG/CR-2989, July 1983.

4. Electric Power Research Institute, “The Reliability of Emergency Diesel at U.S. Nuclear Power Plants,” NSAC-108, September 1986.

5. U.S. Nuclear Regulatory Commission, “Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, ‘Station Blackout’,” NUREG-1109, June 1988.

6. Science and Engineering Associates, Inc., and Mathtech, Inc., “Response To Industry Comments on Station Blackout Cost Analysis (NUREG/CR-3840),” November 1986.

7. Nuclear Energy Institute, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,” NUMARC 87-00, Rev. 0, November 1987.

8. U.S. Nuclear Regulatory Commission, “Final Station Blackout Rule, USI A-44,” SECY-88-22, January 21, 1988.

9. U.S. Nuclear Regulatory Commission, “Resolution of Generic Safety Issue B-56, Diesel Generator Reliability,” SECY-93-044, February 22, 1993.

10. Nuclear Energy Institute, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,” NUMARC 87-00, Rev. 1, August 1991.

11. Nuclear Managment and Resources Council, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” NUMARC 93-01, May 1993.

12. U.S. Nuclear Regulatory Commission, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Regulatory Guide 1.160, June 1993.

13. U.S. Nuclear Regulatory Commission, “Inspection of Implementation of Station Blackout Rule,“ Temporary Instruction 2515/120, no date.

14. U.S. Nuclear Regulatory Commission, Inspection Procedure 62706, “Maintenance Rule,” December 31, 1997.

APPENDIX B

PLANT-SPECIFIC STATION BLACKOUT INFORMATION BY REACTOR TYPE AND OPERATING STATUS

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-1 Operating Pressurized-Water Reactors

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Arkansas Nuclear One Unit 1	4.67E-05	1.58E-05	33.8	4/.95/10/1	Added 1 DG and crosstie	3.58E-02	2	1			
Arkansas Nuclear One Unit 2	3.40E-05	1.23E-06	3.6	4/.95/10/1	Added crosstie	5.84E-02	1	1			
Beaver Valley Unit 1	2.14E-04	6.51E-05	30.4	4/.975/60/1	Added crosstie	6.64E-02	2				
Beaver Valley Unit 2	1.92E-04	4.86E-05	25.3	4/.975/60/1	Added crosstie	7.44E-02	1				
Braidwood Units 1&2	2.74E-05	6.20E-06	22.6	4/.95/10/1		4.53E-02	2				
Bryon Units 1&2	3.09E-05	4.30E-06	13.9	4/.95/10/1		4.43E-02					
Callaway	5.85E-05	1.80E-05	30.8	4/.975/-/1		4.60E-02					
Calvert Cliffs Units 1&2	2.40E-04	8.32E-06	3.4	4/.975/60/4	Added 1 EDG and one 1 DG	1.36E-01	3				
Catawba Units 1&2	5.80E-05	6.0E-07	10.3	4/.95/10/1		2.0E-03	1			330	
Comanche Peak Units 1&2	5.72E-05	1.5E-05	26.2	4/.95/-/1							

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Crystal River Unit 3	1.53E-05	3.28E-06	21.5	4/.975/-/4	dc load shed. Added nonclass 1E battery	4.35E-01	3				
Davis-Besse	6.6E-05	3.50E-05	53	4/.95/10/2	Added 1 DG	3.50E-02	2	1		1680	
DC Cook Units 1&2	6.2E-05	1.13E-05	18.1	4/.975/-/2	dc load shed	4.0E-02	1				
Diablo Canyon Units 1&2	8.8E-05	5.0E-06	5.68	4/.95/-/1	Added 1 DG	9.1E-02	1				261 917
Farley Units 1&2	1.3E-04	1.22E-05	9.4	4/.95/10/3	Service water to Aac, auto load shedding	4.70E-02	2				
Fort Calhoun	1.36E-05	NA	–	4/.95/-/2	DC load shed	2.17E-01	2				
Ginna	8.74E-05	1.0E-06	1.14	4/.975/-/1		3.50E-03	4				
Harris	7.0E-05	1.71E-05	24.4	4/.95/-/3	Lighting in several areas, ladder to isolation valve						
Indian Point Unit 2	3.13E-05	4.47E-06	14.3	8/.95/60/2	Added a DG for gas turbine auxiliaries	6.91E-02	2		3	390	

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Indian Point Unit 3	4.40E-05	4.80E-06	10.9	8/.95/60/2		6.80E-02	1				
Kewaunee	6.6E-05	2.64E-05	40	4/.95/60/2	Cross-tie to nonsafety power source	4.4E-02					
McGuire Units 1&2	4.0E-05	9.26E-06	23.3	4/.95/10/1		7.0E-02	3				
Millstone Unit 2	3.42E-05	1.0E-10	NMN	8/.975/60/5	Upgraded unit 1-2 crosstie	9.10E-02	1	1		330	
Millstone Unit 3	5.61E-05	5.10E-06	6	8/.975/60/5	Added DG	1.12E-01					
North Anna Units 1&2	7.16E-05	8.0E-06	11.2	4/.95/60/4	Added DG, switchgear, crosstie	1.14E-02					
Oconee Units 1, 2&3	2.3E-05	2.57E-06	11.2	4/.975/10/1		9.0E-02	2				
Palisades	5.07E-05	9.10E-06	17.9	4/.95/-/1	DC load shed, compressed air for ADVs	3.0E-02	3			388	
Palo Verde Units 1, 2&3	9.0E-05	1.91E-05	21.2	4/.95/10/2	Added 2 gas turbines	7.83E-02	3			1138	
Point Beach Units 1&2	1.15E-04	1.51E-05	13.1	4/.975/60/2	Gas turbine modifications	6.10E-02	4				

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Prairie Island Units 1&2	5.05E-05	3.1E-06	6.14	4/.975/10/3	Added 2 EDGs	-	1	2		296 296	
Robinson Unit 2	3.20E-04	2.6E-05	8.13	8/.95/60/4	Modified conduit supports in switchgear room	6.1E-02	2			454	
Salem Unit 1	5.20E-05	2.10E-05	40.4	4/.975/-/2	EDG compressed air mod	6.0E-02	1				
Salem Unit 2	5.5E-05	1.70E-05	30.9	4/.975/-/2	EDG compressed air mod	6.0E-02	2			655	1675
San Onofre Units 2&3	3.0E-05	2.0E-06	6.67	4/.95/-/1	DC load shed and crosstie	1.1E-01			2		
St. Lucie Unit 1	2.30E-05	2.65E-06	11.5	4/.975/10/5	Added crosstie	1.5E-01	1		3		
St. Lucie Unit 2	2.62E-05	2.64E-06	10.1	4/.975/10/5	Added crosstie	1.5E-01					
Seabrook	6.86E-05	1.53E-05	22.3	4/.975/-/3	DC load shed	4.93E-02					
Sequoyah Units 1&2	1.70E-04	5.32E-06	3.2	4/.975/-/2	DC load shed, added air supply	5.16E-03	2				

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-1 Operating Pressurized-Water Reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Summer	2.0E-04	4.9E-05	24.5	4/.95/-/3	DC load shed, battery mod	7.3E-02			1		
South Texas Units 1&2	4.3E-05	1.46E-05	34.9	4/.975/10/5	Procedural cross-tie						
Surry Units 1&2	1.25E-04	8.09E-06	6.47	4/.975/10/4	Added DG	7.69E-02					
Three Mile Island Unit 1	4.49E-04	1.57E-05	3.5	4/.975/10/3	Modifications to existing DGs	5.68E-02					
Turkey Point Units 3&4	3.73E-04	4.70E-06	1.2	8/.95/10/5	Added 2 EDGs and cross-tie	1.7E-01	4	2	7	7950 7908	335
Votgle Units 1&2	4.9E-05	4.4E-07	11	4/.95/-/2	Added 5 circuit breakers and lighting	6.6E-04					
Waterford Unit 3	1.80E-05	6.24E-06	34.7	4/.975/-/4	DC load shed. Added portable air compressors for EDGs	3.6E-02					
Watts Bar Unit 1	8.0E-05	1.73E-05	21.6	4/.975/-?/1		3.64E-02					
Wolf Creek	4.2E-05	1.88E-05	44.8	4/.95/-/1		5.12E-02					

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-2 Operating Boiling-Water Reactors

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Browns Ferry Units 2&3	4.80E-05	1.30E-05	27	4/.95/-/1	dc load shed	1.12E-01					
Brunswick Units 1&2	2.70E-05	1.80E-05	66.7	4/.975/60/5	Modified controls for existing crosstie	7.40E-02	3				1508 814
Clinton	2.66E-05	9.8E-06	36.8	4/.95/10/1	Added gas fans for selected room cooling	8.40E-02					
Cooper	7.97E-05	2.77E-05	34.8	4/.95/-/2		3.50E-02					
Dresden Units 2&3	1.8E-05	9.30E-07	5.03	4/.95/60/2	Added 2 DGs	1.12E-01	3	1		240	
Duane Arnold	7.84E-06	1.90E-06	24.2	4/.975/-/2	dc load shed, RCIC insulation & main control room lighting	1.17E-01			1		
Fermi	5.70E-06	1.3E-07	NMN	4/.95/60/1		1.88E-01					
FitzPatrick	1.92E-06	1.75E-06	NMN	4/.95/-/1	dc load shed, instrumentation and power supply mods	5.70E-02					
Grand Gulf	1.77E-05	7.46E-06	36.8	4/.95/-/2	dc load shed	6.80E-02					

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-2 Operating Boiling-Water Reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Hatch Unit 1	2.23E-05	3.30E-06	14.8	4/.95/60/2	Replaced battery chargers	2.20E-02					
Hatch Unit 2	2.36E-05	3.23E-06	13.7	4/.95/60/2	Replaced battery chargers	2.20E-02					
Hope Creek	4.63E-05	3.38E-05	73	4/.95/-/2	Valve modifications	3.4E-02					
LaSalle Units 1&2	4.74E-05	3.82E-05	80.6	4/.975/-/1	dc load shed , New batteries	9.60E-02	1				
Limerick Units 1&2	4.30E-06	1.0E-07	NMN	4/.95/60/3	Upgraded cross-ties	5.9E-02					
Monticello	2.60E-05	1.20E-05	46.2	4/.95/-/1	dc load shed	7.90E-02					
Nine Mile Point Unit 1	5.50E-06	3.50E-06	NMN	4/.975/-/1	dc load shed, added two safety related batteries	5.00E-02	4			595	
Nine Mile Point Unit 2	3.10E-05	5.50E-06	17.7	4/.975/-/1	dc load shed	1.20E-01					

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-2 Operating Boiling-Water Reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Oyster Creek	3.90E-06	2.30E-06	NMN	4/.975/60/1	Added crosstie & reactor pressure indication	3.26E-02	3				240
Peach Bottom Units 2 & 3	5.53E-06	4.81E-07	8.7	8/.975/60/3	Cross-tie to hydro unit	5.9E-02					
Perry	1.30E-05	2.25E-06	43.4	4/.95/10/1	Replaced selected cables	6.09E-02					
Pilgrim	5.80E-05	1.0E-10	NMN	8/.975/10/4	Alarms to line-up Aac	6.17E-01	1	5			1263 534
Quad Cities Units 1&2	1.2E-06	5.72E-07	NMN	4/.95/60/1	Added 2 DGs	4.81E-02	2				
River Bend	1.55E-05	1.35E-05	87.5	4/.95/-/2	Minor structural mod	3.50E-02	1				
Susquehanna Units 1&2	1.7E-05	4.2E-11	NMN	4/.975/-/2	dc load shed	-	1				
Vermont Yankee	4.30E-06	9.17E-07	21.3	8/.975/10/4	Modified incoming line and controls	1.0E-01	2			277	
Washington Nuclear Plant Unit 2	1.73E-05	1.07E-05	61.1	4/.95/-/1	dc load shed, replaced inverters	2.46E-02					

Plant-Specific Station Blackout Information By Reactor Type and Operating Status

Table B-3 Reactors No Longer Operating

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Big Rock Point	5.40E-05	5.10E-07	NMN	4/.95/-/1	DC load shed, added crosstie	2.8E-01					
Browns Ferry Unit 1	4.80E-05	1.30E-05	27	4/.95/-/1		1.12E-01					
Haddam Neck	1.90E-04	8.70E-06	4.46	4/.95/-/2	Fuel system for gas water pump	9.0E-02	5				
Maine Yankee	7.40E-05	1.11E-05	15	4/.975/60/3		5.0E-02	1				
Millstone Unit 1	1.13E-05	7.00E-06	62	8/.975/60/5	Upgraded crosstie		1	2		300	
Zion Units 1&2	4.0E-06	4.4E-07	NMN	4/.95/10/1		4.60E-02	1				

APPENDIX C

COMPARISON OF SELECTED STATION BLACKOUT CHARACTERISTICS

Comparison of Selected Station Blackout Characteristics

PLANT	LOOP Initiating Frequency			Extreme Severe Weather Group 5	Selected SBO characteristics
	PRA/IPE	Actual frequency/number of LOOP events at power since commercial operation	Approximate factor PRA/IPE underestimated in comparison to operating experience		SBO-CDF/Coping time in minutes/ Aac access time in minutes
Pilgrim	6.17E-01	1.48E-01/4			E-10/8/10
Crystal River	4.35E-01	1.36E-01/3			E-06/4/-
Fort Calhoun	2.17E-01	0.76E-01/2			E-06/4/-
Fermi 2	1.88E-01	0/0			E-07/4/60
Turkey Point 3 Turkey Point 4	1.7E-01 1.7E-01	5.18E-01/14 0.76E-01/2	3	X X	E-06/8/10 E-06/8/10
St. Lucie 1 St. Lucie 2	1.5E-01 1.5E-01	1.73E-01/4 0/0		X X	E-06/4/10 E-06/4/10
Calvert Cliffs 1 Calvert Cliffs 2	1.36E-01 1.36E-01	0.4E-01/1 0.43E-01/1			E-06/4/60 E-06/4/60
South Texas 1&2	1.32E-01	0/0		X	E-05/8/10
Nine Mile 2	1.2E-01	0/0			E-06/4/-
Duane Arnold	1.17E-01	0.4E-01/1			E-06/4/-
Browns Ferry 2	1.12E-01	0/0			E-05/4/-
Dresden 2 Dresden 3	1.12E-01 1.12E-01	1.07E-01/3 0.35E-01/1			E-07/4/60
Millstone 3	1.12E-01	0/0		X	E-06/8/60
San Onfre 2&3	1.1E-01	0/0			E-06/4/-
Vermont Yankee	1.0E-01	0.38E-01/1			E-07/8/10
Millstone 2	9.1E-02	12.5E-02/3	2	X	E-10/8/60
Palo Verde 1	7.83E-02	15.3E-02/2			E-05/4/10
Brunswick 1 Brunswick 2	7.4E-02 7.4E-02	9.1E-02/2 0/0	2	X X	E-05/8/60 E-05/8/60
Indian Point 2 Indian Point 3	6.91E-02 6.80E-02	20E-02/5 4.3E-02/1	3		E-06/8/60 E-06/8/60
Robinson	6.1E-02	7.1E-02/2			E-05/8/60
Point Beach 1 Point Beach 2	6.0E-02 6.0E-02	10.3E-02/3 3.7E-02/1			E-05/8/60 E-05/8/60
Salem 2	6.0E-02	11.1E-02/2	2		E-05/4/-
Oyster Creek	3.26E-02	10E-02/3	3		E-06/4/60
Ginna	3.50E-03	137E-03/4	39		E-06/4/-
Palisades	3.0E-03	71E-03/2	23		E-06/4/-

APPENDIX D

EMERGENCY DIESEL GENERATOR TRIGGER VALUES

EMERGENCY DIESEL GENERATOR TRIGGER VALUES

1 History of EDG Trigger Failure Rates

Regulatory Guide (RG) 1.155, “Station Blackout,” August 1988, (paragraph 1.1) [Ref. 1], expected that the individual EDG reliability would be calculated in accordance with NSAC-108, “The Reliability of Emergency Diesel at U.S. Nuclear Power Plants,” September 1986 [Ref. 2], as the product of start reliability and the load-run reliability using data that met the NSAC-108 definition for valid EDG start and load-runs. The method provides for the calculation of individual EDG reliablity as shown in NSAC-108 and “A Review of Issues Related to Improving Nuclear Power Plant Diesel Generator Reliability,” NUREG/CR-4557, March 1986 [Ref. 3], which calculated the reliability for each nuclear plant EDG using 3 years of plant EDG data.

RG 1.155 (paragraph 1.2) provides that licensees calculate a unit average EDG reliability for 20, 50, and 100 demands. RG 1.155 (paragraphs 1.1.3, 1.1.4, and 1.1.5) provides for licensees to compare the unit average EDG reliability against reliability criteria for determining the station blackout (SBO) coping duration and the individual EDG target reliability based upon the emergency ac power system configuration (EAC) group and whether any, or none, of the reliability criteria are met.

Subsequently, the notion of calculating EDG reliability based on 20, 50, and 100 demands emerged. Failures were associated with 20, 50, and 100 demands and statistical analyses completed, resulting in various combinations and use of trigger values as summarized below.

– NUREG/CR-5078, “A Reliability Program for Emergency Diesel Generators at Nuclear Power Plants,” February 1988 [Ref. 4], evaluated failure criteria for EDGs having a target reliability of 0.95 or 0.975, based on the number of failures recorded in the last 20, 50, and 100 demands using a Monte Carlo simulation. NUREG/CR-5078 analyzed eight failure progressions, considering the false alarm rate (a percentage of the time the true EDG reliability is observed), and provided interpretations of the failure progressions. For example, the first failure progression for the 0.95 target reliability group is an unacceptable, requires-immediate-attention condition when the observed performance is 2 failures in 20 demands, 5 failures in 50 demands, and 10 failures in 100 demands.

– NUREG/CR-5611, “Issues and Approaches for Using Equipment Reliability Alert Levels,” June 1991 [Ref. 5], determined the false-alarm rate and the detection response of seven candidate alert systems provided by the industry to a specified reliability degradation from .98 to 0.92 for different criteria supplied by the industry.

– The Commission endorsed an approach to the resolution of generic safety issue B-56 in SECY-90-340, “Diesel Generator Reliability B-56, Resolution of Generic Safety Issue 56, (COMJC-91-001/001-A0),” June 28, 1991 [Ref. 6], that had the following elements: (1) target reliability levels would be established for each licensee’s EDGs (consistent with the approach in the maintenance rule) and (2) trigger values would be used to provide an “early warning” of EDG degradation and to provide a basis for taking regulatory action and a reporting regime would be established in accordance with the approach specified above. SECY-90-340 also specifies that the trigger values would be used to (1) report diesel failures to the NRC when such failures reach 3 failures out of

20 starts and (2) undertake accelerated testing and report to the NRC when the failures reach 4 out of 20 failures. "Double trigger" values of 5 failures out of 50 starts (5 in 50) and 8 in 100 for a 0.95 target reliability, and 4 in 50 and 8 in 100 for a 0.975 target reliability would provide clear indication that the specified underlying reliability has not been met.

– An NRC white paper, "A Sequential Trigger Procedure for Use in Monitoring Nuclear Power Plant Emergency Diesel Generator Reliability," October 1992 [Ref. 7], used a Monte Carlo simulation to evaluate the following triggers: 3 in 20 as an early warning for individual and all EDGS; 4 in 25 as a problem EDG; and double triggers of 5 in 50 and 8 in 100 for a 0.95 target reliability or 4 in 50 and 5 in 100 for an 0.975 EDG target reliability.

– NUMARC (now NEI) developed an EDG reliability program in Appendix D of NUMARC 87-00, Rev. 1, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light-Water Reactors," August 1991 [Ref. 8], for the maintenance rule. Appendix D of NUMARC 87-00, Rev.1 (paragraph D.2.3.1), employs the use of trigger values shown in Table E-1, "Industry Emergency Diesel Generator Trigger Failure Rates," to indicate when EDGs do not meet the selected target reliabilities. The trigger values represent the point at which additional actions should be taken to restore the selected EDG reliability. NUMARC 87-00, Rev. 1, prescribes actions for reaching one trigger value and another set of actions for reaching the triggers corresponding to 50 and 100 demands.

Table D-1 Industry Emergency Diesel Generator Trigger Failure Rates

	Selected Target Reliability	Failures In 20 Demands	Failures in 50 Demands	Failures in 100 Demands
NUMARC 87-00, Rev. 1, Appendix D	0.95	3	5	8
	0.975	3	4	5

– In a letter to the ACRS dated October 29, 1993, the Executive Director for Operations advised that the staff agreed with the ACRS that conformance of individual EDGs with trigger values cannot be taken in any statistical fashion and that the EDG has demonstrated achievement of licensee high reliability value commitments of 0.95 or 0.975. The October letter states that Note 3 was added to RG 1.160, dated June 1993, to emphasize this fact. Note 3 states that the triggers are intended to indicate when EDG performance problems exist such that additional monitoring or corrective action is necessary and that it is not practical to demonstrate by statistical analysis that conformance to the trigger values will ensure the attainment of high reliability, with a reasonable degree of confidence, of individual EDG units.

The staff has not endorsed NUMARC 87-00, Rev. 1. Past and present revisions of RG 1.160 (Section C), state that NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," May 1993 [Ref. 9], (which

references NUMARC 87-00, Rev. 1, and uses the triggers in an example in Section 12.2.4) references other documents, but NRC's endorsement of NUMARC 93-01 should not be considered an endorsement of the referenced documents. The NRC staff intended that this statement be interpreted to mean NUMARC 87-00, Rev. 1, was not endorsed. However, as explained in Appendix E, the wording in past revisions of RG 1.160 from 1993 and 1997 could lead to the conclusion that the trigger values could be used to demonstrate compliance with the EDG reliability commitments.

– The wording in past revisions of RG 1.160 could lead to the conclusion that the NUMARC reference documents does not apply to the EDGs. For example, RG 1.160, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Regulatory Guide 1.160, June 1993 (Section C) [Ref. 10], also states that the example in NUMARC 93-01, (Section 12.2.4), describes an acceptable method for establishing EDG performance criteria and/or goals and subsequently monitor EDG performance. As another example RG 1.160, Rev. 1, January 1995, (Section B) [Ref. 11], states that the EDGs are required to be handled under paragraphs (a)(1) and (a)(2) of the maintenance rule as described in NUMARC 93-01.

– Inspection documents Temporary Instruction 2515/120, "Inspection of Implementation of Station Blackout Rule," (Section 2515/120-040) [Ref. 12], and Inspection Procedure 62706, "Maintenance Rule," December 31, 1997, (Section 3.05) [Ref. 13], use the NUMARC 87-00, Rev. 1, Appendix D, trigger values for assessing compliance with the RG 1.155 minimum individual EDG target reliabilities of 0.95 and 0.975.

– A search of licensing correspondence indicates that several licensees may be using the NUMARC trigger values to demonstrate compliance to the RG 1.155 EDG target reliabilities.

2 Technical Basis of the Trigger Values From Different Points of View

The ACRS challenged the trigger statistic using confidence intervals developed from the binomial distribution and from the point of view that trigger values do not provide sufficient statistical confidence that the target reliabilities are met. INEEL was requested to develop statistics, consistent with the ACRS point of view, uses the binomial distribution to indicate the confidence in meeting or exceeding a reliability given failure data. In addition, INEEL was asked to develop statistics using Bayesian methods to provide the probability that the reliability is greater than the target reliability, given the failure data. INEEL provided the following results and Table D-2, "Statistical Confidence That A Reliability Is Met," below.

INEEL was asked to investigate the statistical significance of industry emergency diesel generator (EDG) trigger failure rates in ensuring that EDG reliability remains above target reliability levels (0.95 for some plants and 0.975 for others). Table E-2, "Statistical Confidence That EDG Reliability Is Met," shows the results of the INEEL investigation. The result of this investigation is that when EDG failure rates reach the trigger levels represented by the bold data in the table below, the probability that their reliability meets or exceeds the associated target reliabilities is, at best, less than 6 percent. The discussion below provides more detail about this finding.

Table D-2
Statistical Confidence That Emergency Diesel Generator Reliability Is Met

Failure Data		Estimated reliability	Binomial distribution lower confidence bounds on estimated reliability				Probability that reliability > target reliability	
Failures	Demands		2.5%	5%	50%	95%	Target reliability 0.95	Target reliability 0.975
2	20	0.90	0.968	0.958	0.869	0.717	0.029	0.021
3		0.85	0.943	0.929	0.819	0.656	**0.056**	**0.044**
4		0.80	0.913	0.896	0.770	0.599	0.087	0.072
3	50	0.94	0.978	0.972	0.927	0.852	0.022	0.017
4		0.92	0.967	0.960	0.907	0.826	0.034	**0.028**
5		0.90	0.955	0.946	0.887	0.801	**0.047**	0.039
6		0.88	0.942	0.932	0.867	0.777	0.061	0.052
4	100	0.96	0.984	0.980	0.953	0.911	0.017	0.014
5		0.95	0.978	0.974	0.943	0.898	0.023	**0.019**
6		0.94	0.971	0.967	0.934	0.885	0.030	0.025
7		0.93	0.965	0.960	0.924	0.873	0.037	0.032
8		0.92	0.958	0.952	0.914	0.860	**0.044**	0.039
9		0.91	0.951	0.945	0.904	0.848	0.051	0.045
10		0.90	0.944	0.937	0.894	0.836	0.059	0.053

Notes:
Bold entries correspond to the trigger failure rate levels.
Estimated reliability: Successes/Demands.
Lower confidence bounds: Based on data from a binomial distribution. The reliability is greater than the values cited, with the confidences stated in the column headings.
Probability of exceeding specified targets: computed as probability that a beta random variable with parameters (Successes+0.5, Failures+0.5) equals or exceeds the specified target.

In the Table D-2 results for various sets of failure, demand combinations surrounding the Trigger Failure Rates (in bold) are provided. The confidence bound section of the table gives examples of the level of information available from each set of data about the probability of success, (i.e., the reliability, using a binomial distribution). For example within 20 demands and 2 failures, the confidence in exceeding a 0.958 reliability (between the two target reliabilities) is only 5 percent . The confidence section shows that with three failures in 20 demands, the confidence in meeting or exceeding a reliability of 0.95 is less than 2.5 percent. In the right-most lower bound column where a relatively high (95 percent) exceedance confidence exists, the reliabilities are considerably less than the target reliabilities.

The last two columns in the table contain results for the probability of meeting or exceeding the specific target reliabilities given each set of data. The calculations are based on a probability distribution for the success probability for each set of data, developed using the "simple Bayes" method (see G. Box and G Tiao, *Bayesian Inference in Statistical Analysis*, Reading, MA: Addison Wesley, 1973, Sections 1.3.4-1.3.5). This method is widely used in risk assessment; for example, it is described in NUREG/CR-2300 (the 1983 PRA Procedures Guide published by the NRC). A beta distribution is a conjugate prior distribution for binomially-distributed data. In the simple Bayes method, a noninformative prior beta distribution (called the Jeffreys prior) is updated with observed data, resulting in a posterior beta distribution that describes the reliability for each set of data (each row in the table). The probabilities given in the table are tail probabilities for the resulting distributions. The probabilities highlighted correspond to the trigger failure data. All of these probabilities are less than 0.06.

From this perspective, the trigger failure rates do not ensure the RG 1.155 reliability levels with a reasonable degree of confidence.

From an NRC inspection point of view, action should only be taken if there is high confidence that the reliability is NOT maintained. INEEL was also asked to develop statistics using the binomial distribution to indicate the confidence that the reliability is NOT as indicated given given failure data. INEEL provided the following results and Table D-3, “Statistical Confidence That a Reliability Is NOT Met” below.

INEEL was also asked to provide the same type of information from the point of view that action should be taken only if there is high confidence that the reliability is NOT maintained [which] is provided below. The results are shown in Table E-3, “Statistical Confidence That EDG Reliability Is NOT Met” to demonstrate use and interpretation of the information, an example follows: after 2 failures in 20 demands, we are 95 percent confident that the reliability is less than 98 percent; but when this behavior continues and we see 10 failures in 100 demands, we are 95 percent confident that the reliability is less than 94.5 percent. We are more confident that the reliability is inadequate, and that changes are needed, after more failures are seen.

TABLE D-3
Statistical Confidence That Emergency Diesel Generator Reliability Is NOT Met

Failure Data		Estimated reliability	Binomial distribution upper confidence bounds on estimated reliability				Probability that reliability < target reliability	
Failures	Demands		50%	90%	95%	97.5%	Target reliability 0.95	Target reliability 0.975
2	20	0.90	0.917	0.973	0.982	0.988	0.853	0.965
3		0.85	0.869	0.944	0.958	0.968	**0.964**	**0.996**
4		0.80	0.819	0.910	0.929	0.943	0.993	1.000
3	50	0.94	0.947	0.978	0.983	0.987	0.660	0.929
4		0.92	0.927	0.965	0.972	0.978	0.839	**0.982**
5		0.90	0.907	0.951	0.960	0.967	**0.936**	0.996
6		0.88	0.887	0.936	0.946	0.955	0.979	0.999
4	100	0.96	0.963	0.982	0.986	0.989	0.344	0.837
5		0.95	0.953	0.975	0.980	0.984	0.528	**0.934**
6		0.94	0.943	0.968	0.974	0.978	0.696	0.977
7		0.93	0.934	0.961	0.967	0.971	0.825	0.993
8		0.92	0.924	0.953	0.960	0.965	**0.909**	0.998
9		0.91	0.914	0.945	0.952	0.958	0.957	1.000
10		0.90	0.904	0.937	0.945	0.951	0.982	1.000

Notes:
Bold entries correspond to the trigger failure rate levels.
Estimated reliability: Successes/Demands.
Upper confidence bounds: Based on data from a binomial distribution. The reliability is less than the values cited, with the confidences stated in the column headings.
Probability of being below specified targets: computed as probability that a beta random variable with parameters (Successes+0.5, Failures+0.5) is less than the specified target.

From this perspective, the conclusion is the trigger values ensure, with a high degree of confidence, that the EDG is NOT being met. On one hand these failure rates can be used to ensure that licensees or NRC inspectors do not

take action when no fault exists. On the other hand, the time between the onset of degradation of EDG performance and the sure detection can be long. During this time, the plant could be in a vulnerable position, in which the underlying risk of an SBO could have risen to an unacceptable level; an effective EDG reliability program should anticipate this condition.

REFERENCES

1. U.S. Nuclear Regulatory Commission, “Station Blackout,” Regulatory Guide 1.155, August 1988.

2. Electric Power Research Institute, “The Reliability of Emergency Diesel at U.S. Nuclear Power Plants,” NSAC-108, September 1986.

3. U.S. Nuclear Regulatory Commission, “A Review of Issues Related to Improving Nuclear Power Plant Diesel Generator Reliability,” NUREG/CR-4557, March 1986.

4. U.S. Nuclear Regulatory Commission, “A Reliability Program for Emergency Diesel Generators at Nuclear Power Plants,” NUREG/CR-5078, February 1988.

5. U.S. Nuclear Regulatory Commission,“Issues and Approaches for Using Equipment Reliability Alert Levels,” NUREG/CR-5611, June 1991.

6. U.S. Nuclear Regulatory Commission, “Diesel Generator Reliability B-56, Resolution of Generic Safety Issue 56, (COMJC-91-001/001-A0),” SECY-90-340, June 28, 1991.

7. Los Alamos National Laboratory, “A Sequential Trigger Procedure For Use In Monitoring Nuclear Power Plant Emergency Diesel Generator Reliability,” October 1992.

8. Nuclear Energy Institute, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,” NUMARC 87-00, Rev. 1, August 1991.

9. Nuclear Management and Resources Council, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” NUMARC 93-01, May 1993

10. U.S. Nuclear Regulatory Commission, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Regulatory Guide 1.160, June 1993.

11. U.S. Nuclear Regulatory Commission, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” RG 1.160, Rev. 1, January 1995.

12. U.S. Nuclear Regulatory Commission, “Inspection of Implementation of Station Blackout Rule,“ Temporary Instruction 2515/120, no date.

13. U.S. Nuclear Regulatory Commission, “Maintenance Rule,” IP 62706, December 31, 1997.

APPENDIX E

STATION BLACKOUT RULE
ACTIVITY AND MODIFICATION SUMMARY

Table E-1 Station Blackout Rule Activity and Modification Summary

Modifications credited in the safety analysis	Expected Estimated Cost Impact	Outcomes	
		Plant Name	Estimated Cost Impact To Implement
Coping analysis, procedures, and training	$35M (100 plants at $350K each)		$37.8M (108 plants at $350K each)
Excess capacity EDG			
–From existing EDG		Braidwood 1 & 2, Browns Ferry 1, 2&3, Byron 1&2	0
–From existing EDG but added or upgraded cross-tie		Beaver Valley 1&2, Brunswick 1&2, Limerick 1&2, Millstone 1&2, St. Lucie 1&2	$1.0M (5 modifications at $200K each)
–Added one EDG		Calvert Cliffs 1&2 (not excess capacity), Diablo Canyon 1&2	$20M (2 modifications at $10M each)
–Added two EDG		Prairie Island 1&2, Turkey Point 3&4	$40M (2 modifications at $20M each)
Existing excess redundancy EDG		Farley 1&2, Hatch 1&2, South Texas 1&2, Zion 1&2	0
Existing HPCS EDG		Clinton, Perry	0
Non-Class 1E DG			
Added 1 DG		ANO 1&2, Calvert Cliffs 1&2, Davis-Besse, North Anna 1&2, Millstone 3, Surry 1&2	$54M (6 modifications at $9M each)
Added 2 DGs		Dresden 2 & 3, Quad Cities 1&2	$30M (2 modifications at $15M each)
Modified existing DG		Kewaunee, Pilgrim, Three Mile Island 1	$600K (3 modifications at $200K each)

Table E-1 Station Blackout Rule Activity and Modification Summary (Cont.)

Modifications credited in the safety analysis	Expected Estimated Cost Impact	Outcomes	
		Plant Name	Estimated Cost Impact To Implement
Non-Class 1E combustion turbine Existing cross-tie Added cross-tie		 Fermi 2 Oyster Creek	 0 $500K
Non-Class 1E gas turbine Modified existing Improve reliability Added two		 Indian Point 2 Point Beach 1&2 Palo Verde 1, 2&3	 $400K $500K $30M
Use hydro generator Existing Added cross-tie		 Vermont Yankee Peach Bottom 2&3	 0 $500K
Appendix R DG		Catawba 1&2, Indian Point 3 Maine Yankee, McGuire 1&2, Oconee 1, 2&3; Robinson 2	0
Battery load shedding required		Big Rock Point, Browns Ferry 1, 2&3, Crystal River, DC Cook 1&2, Duane Arnold, FitzPatrick, Fort Calhoun, Grand Gulf, , LaSalle 1&2, Nine Mile Point 1&2, Monticello, Palisades, Salem 1&2, San Onfre 2&3, Seabrook 1, Sequoyah 1&2, Summer, Susquehanna 1&2, Washington Nuclear Power 2, Waterford 3	$2.8M (28 procedure modifications at $100K each)
Improve EDG reliability	$2.5M (10 modifications)		

Table E-1 Station Blackout Rule Activity and Modification Summary (Cont.)

Modifications credited in the safety analysis	Expected Estimated Cost Impact	Outcomes	
		Plant Name	Estimated Cost Impact To Implement
Requalify an EDG	$5.6M (2 modifications)		
Minor mods to cope	$11.9M (17 minor modifications	Brunswick 1&2, Clinton, Duane Arnold, FitzPatrick, Haddam Neck, Harris, Farley 1&2, Hope Creek, Monticello, Oyster Creek, Palisades, Palo Verde 1, 2&3; Perry, Pilgrim, River Bend, Robinson 2, Salem 1&2, South Texas, Vogtle 1&2	$12M
DC system modifications to cope	$5M (10 modifications at $500K each)		$5M (10 modifications at $500K each)
Added batteries		LaSalle 1&2 , Nine Mile Point 1	
Added non-Class IE battery		Crystal River	
Add/replace battery chargers		Hatch 1&2, Summer	
Added cross-tie		San Onfre 2&3	
Replace inverters		WNP-2	
TOTAL	$60M		$235M

APPENDIX F

OPERATING EVENTS

Operating Events

Table F-1 Losses of Offsite Power Since 1990 Having Recovery Times Greater Than 4 Hours

Plant **Reference Document** **Event Date** **Summary of LOOP Event and Reactor Status**	**Description of the Event**
Davis-Besse LER 346/98-006 June 24, 1998 Tornado and near SBO event ASP Report for LER 346/98-006 February 1999	A tornado damaged the Davis-Besse switchyard and caused a LOOP for approximately 28 hours (1690 minutes). The EDGs were both manually started upon report of a tornado; however, one EDG failed to start from the control room and was successfully started locally. During the event, one EDG was technically inoperable since the tornado damaged a roof mounted room cooling and resulted in slightly elevated room temperatures. The tornado caused significant damage to the Ottawa County electrical distribution system, making 40 percent of the sirens inoperable. There were several equipment malfunctions that were either successfully addressed by operations or negligible. Post-event analysis as part of the ASP Program identified that when the SBO-DG is in standby, a nonessential bus supplies power to the SBO-DG. If the nonessential bus is not powered, then the batteries will deplete in approximately 20 hours. Had the EDGs failed to start or run for the first 8 hours of the 28-hour event, the SBO-DG may have been unavailable and this could have led to core damage.
Prairie Island 1 & 2 LER 282/96-012 June 29, 1996 Weather-related LOOP while at power	High winds caused a unit trip and LOOP on both units, lasting 296 minutes.
Catawba 2 LER 414/96-001 February 6, 1996 Plant centered LOOP while at power. Near SBO event	Failed insulators in the isolated phase bus duct caused a LOOP for 330 minutes. One of two EDGs was inoperable due to battery charger repairs. The effort to restore power was delayed due to procedural inadequacy.

Table F-1 Losses of Offsite Power Since 1990 Having Recovery Times Greater Than 4 Hours (Cont.)

Plant **Reference Document** **Event Date** **Summary of LOOP Event and Reactor Status**	**Description of the Event**
Salem 2 LER 311/94-007 April 11, 1994 Plant-centered LOOP while at power	A technician's error during testing resulted in a LOOP lasting 385 minutes. No detail was provided about the recovery; however, it appears that extensive trouble shooting to permit adequate assessment delayed power restoration.
Turkey Point 3 &4 LER 250/92-009 August 24, 1992 Weather-related LOOP while at power, near SBO event.	A hurricane, with winds up to 145 mph, passed directly over the site causing a LOOP lasting approximately 6.5 days. The 5 on-site black-start DGs were unavailable due to moisture that caused loss of switchgear between the black-start DGs and the plant safety buses.
Supplemental Report March 1993	The plant modifications completed for SBO rule were heavily relied upon to recover from the event. As a result of the SBO rule, the licensee added two safety EDGs and completed extensive modifications to the Unit 3 and 4 electrical distribution system. A supplemental report, "Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20–30, 1992," shows that during the event one of the two original 2850 kW EDGs failed to run for approximately 2.5 hours and the remaining 3 EDGs carried a load of approximately 3400 kW. Had the SBO rule modifications not been completed, the 3400 kW load would have been in excess of the 2850 kW rating of the remaining EDG.
H.B. Robinson LER 261/92-017 August 22, 1992 Plant-centered LOOP while at power	A LOOP lasting 454 minutes was caused by the failure of a transformer that caused a loss of voltage to one bus and the failure of another transformer to transfer its load that caused a loss of voltage to the remaining bus. The transformer test and repairs account for the duration of the LOOP.

Table F-1 Losses of Offsite Power Since 1990 Having Recovery Times Greater Than 4 Hours (Cont.)

Plant **Reference Document** **Event Date** **Summary of LOOP Event and Reactor Status**	**Description of the Event**
Vermont Yankee LER 271/91-009 April 23, 1991 Plant-centered LOOP while at power	A fault in the switchyard caused a LOOP for 277 minutes. Power restoration was delayed as a result of the length of time required for New England Power Service relay technicians to travel to Vermont Yankee from Providence, Rhode Island, and communication problems between Vermont Yankee and the New England Switching Authority concerning priorities over circuit breaker testing.
Zion 2 LER 304/91-002 March 21, 1991 Plant-centered LOOP while at power. Near SBO event.	During a surveillance test of the firewater system with the unit at power, the deluge valves were inadvertently opened and sprayed water on the main and auxiliary station transformers. This resulted in a generator trip and a LOOP. An EDG was out of service for maintenance. Offsite power was restored in 60 minutes.
Nine Mile Point 1 LER 220/90-023 November 12, 1990 Plant-centered LOOP while at power	The 115 kV cable drop to the reserve transformer that supplies power to the emergency buses broke because of metal fatigue. A phase imbalance resulted in loss of both reserve transformers, a LOOP lasting 595 minutes. Power was restored through one reserve transformer following inspections to assure there was no damage.

Operating Events

Table F-2 Station Blackout Challenges

Plant **Reference Document** **Event Date** **Summary of Event**	**Description of the Event**
St. Lucie 1 & 2 LER 335/98-007 June 30, 1998 SBO challenge	While investigating an anomaly with a Unit 2 sumulator SBO recovery exercise, the licensee discovered that one of the methods to restore electrical power could not be performed as the procedure was written and led to an unanticipated action that could complicate recovery from SBO condition. The cause of the conditions was inadequate SBO recovery procedures that were not verified as part of the modification that installed the SBO cross tie-between the units.
Davis-Besse LER 346/98-006 June 24, 1998 Tornado and near SBO event	See Table D-1 for a description
Indian Point 2 LER 247/98-007 May 22, 1998 SBO AAC unavailability	On 5/22/98, during testing of the Aac system, the licensee determined that the gas turbine No. 3 output circuit breaker was not capable of being closed onto a de-energized bus. On 6/16/98, a similar test determined that the Gas Turbine No. 2 was also not capable of being closed onto a de-energized bus. The cause was an insufficiently comprehensive test for detecting anomalies in the gas turbine control system; previous testing only provided for synchronizing the gas turbine output circuit breaker to an energized bus. The failure of the circuit breakers to close to a de-energized bus was due to the Woodward governor control system. In one case, the control logic was configured for a dead bus closure, and in the other case, the software configuration would not permit dead bus closure.

Table F-2 Station Blackout Challenges (Cont.)

Plant **Reference Document** **Event Date** **Summary of Event**	**Description of the Event**
Millstone 3 & Pilgrim IN 97-21 April 18, 1997 SBO Aac unavailability	IN 97-21, “Availability of Alternate AC Power Source Designed For Station Blackout Event,” was issued to alert addresses to the potential unavailability of an Aac power source during an SBO event at Millstone and Pilgrim as described below. IN 97-21 required the recipients to review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, no specific action or response is required. During inspection activity in 1996, the NRC discovered a potential design deficiency affecting Millstone 3 Aac power source.. The Aac power source includes batteries for (1) a computer that controls and monitors the Aac and (2) Aac dc field flash, oil pump, and breaker control power. The battery chargers for these batteries are fed from offsite power when the AAC power supply is not operating. If Aac power is needed more than 1 hour after the LOOP, the batteries will be so depleted that the Aac-cannot be started and, therefore will not be available if EDGs are lost. On March 7, 1997, the Pilgrim operators attempted to start the SBO-DG 6 hours into a partial LOOP. The SBO DG failed to start because the DG support systems were powered by a nonsafety-related power supply that had been without auxiliary power for an extended period during the LOOP.