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Final Report

Regulatory Effectiveness of the Station Blackout Rule

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ABSTRACT

The Nuclear Regulatory Commission's Office of Nuclear Regulatory Research is reviewing selected regulations, starting with the station blackout (SBO) rule, to determine if the requirements are achieving the desired outcomes. This initiative is part of an evolving program to make NRC activities and decisions more effective, efficient, and realistic. This report evaluates the effectiveness of the SBO rule by comparing regulatory expectations to outcomes. A set of baseline expectations was established from the SBO rule and related regulatory documents in the areas of coping capability, risk reduction, emergency diesel generator reliability, and value-impact. The report concludes that although there are opportunities to improve the clarity of SBO related regulatory documents, the SBO rule is effective and the industry and the NRC costs to implement the SBO rule were reasonable considering the outcome.

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ABBREVIATIONS

Aac	alternate ac
ACRS	Advisory Committee on Reactor Safeguards
AOT	allowed outage time
CDF	core damage frequency
CFR	*Code of Federal Regulations*
EDG	emergency diesel generator
GSI	generic safety issue
IN	information notice
INEEL	Idaho National Engineering and Environmental Laboratory (formerly INEL)
IPE	individual plant examination
LER	licensee event report
LOOP	loss of offsite power
MOOS	maintenance (and testing) out of service
NEI	Nuclear Energy Institute (formerly NUMARC and USCEA)
NRC	Nuclear Regulatory Commission, U.S.
NUMARC	Nuclear Management and Resources Council (now NEI)
PRA	probabilistic risk assessment
RCP	reactor coolant pump
RES	Nuclear Regulatory Research, Office of (NRC)
RG	regulatory guide
RY	reactor-year
SBO	station blackout
SSC	structure, system, and component
USI	Unresolved Safety Issue

EXECUTIVE SUMMARY

The Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (RES) is reviewing selected regulations, starting with the station blackout (SBO) rule, to determine if the regulatory expectations are achieving the desired outcomes. This initiative is part of an evolving program to make NRC activities and decisions more effective, efficient, and realistic. As part of this program, RES is also reviewing the effectiveness of generic safety issue resolution.

The NRC designated SBO, which is a loss of all offsite and onsite ac power concurrent with a turbine trip, as Unresolved Safety Issue A-44 in 1980. In 1988, the Commission concluded that additional SBO regulatory requirements were justified and issued the SBO rule (10 CFR 50.63) to provide further assurance that a loss of both offsite and onsite emergency ac power systems would not adversely affect public health and safety. The SBO rule expected a reduction in the risk as a result of licensees maintaining highly reliable onsite emergency ac electric power supplies; ensuring that the plants can cope with an SBO for some period of time; developing procedures and training to restore offsite and onsite emergency ac power should either become unavailable; and making modifications necessary to meet the SBO rule requirements.

To assess the regulatory effectiveness of the SBO rule, the expectations were established from objective measures as stated in SBO related regulatory documents in the areas of coping capability, risk reduction, emergency diesel generator (EDG) reliability, and value-impact. The outcomes were obtained from realistic information to include the operating experience and NRC equipment reliability studies based on actual safety performance. Comparison of the expectations to the outcomes showed whether the expectations were achieved. Discrepancies between expectations and outcomes prompted a review of the related regulatory documents to find areas that need NRC staff attention. To increase public confidence, earlier drafts of the report were made publicly available and stakeholder comments were openly addressed in an appendix of the report.

The report's conclusion is that the SBO rule was effective considering that the risk expectations were achieved, and that industry and NRC costs to implement the SBO rule were reasonable. In implementing the SBO rule, some plants made hardware modifications (e.g., the addition of diesel generator or gas turbine generator power supplies); and all plants generally maintained EDG reliability at 0.95 or better, and established SBO coping and recovery procedures. Consequently, the plants have gained SBO coping capability, reduced risk, increased the tolerance to a loss of ac offsite or onsite power; and many plants benefitted economically from the addition of power supplies. To elaborate:

- The reduction in the estimated mean SBO core damage frequency (CDF) was approximately 3.2E-05 per reactor year, slightly better than the 2.6E-05 per reactor year expected after implementation of the SBO rule. As a result of the improvements made under the SBO rule, more plants achieved a lower SBO CDF than expected, and the plants with the greatest numbers of loss of offsite power from plant events and extremely severe weather conditions made the most improvement by providing access to an alternate ac power supply. In addition, maintaining high EDG target reliability levels provides assurance that probabilistic risk assessment/individual plant examination EDG performance assumptions are valid. With some exceptions, the observed EDG reliability performance generally exceeds the mean reliability EDG performance assumptions in

the probabilistic risk assessment/individual plant examinations, indicating that SBO CDFs are smaller and better than stated in many probabilistic risk assessment/individual plant examinations. As the SBO rule risk reduction objectives have been exceeded, further investigation of strategies for reducing SBO frequencies (as suggested in NUREG–1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," December 1997) may not be needed.

- As a result of the SBO rule all plants have (1) established SBO coping and recovery procedures; (2) completed training for these procedures; (3) implemented modifications as necessary to cope with an SBO; and (4) ensured a 4- or 8-hour coping capability.

- Before the SBO rule was issued, only 11 of 78 plants surveyed had a formal EDG reliability program, 11 of 78 plants had a unit average EDG reliability less that 0.95, and 2 of 78 had a unit average EDG reliability of less that 0.90. Since the SBO rule was issued, all plants have established an EDG reliability program that has improved EDG reliability. A report shows that only 3 of 102 operating plants have a unit average EDG reliability less than 0.95 and above 0.90 considering actual performance on demand, and maintenance (and testing) out of service (MOOS) with the reactor at power. However, the analysis of EDG performance on demand indicates MOOS with the reactor at power is more than expected and can have a significant effect on the EDG reliability calculations. Increased MOOS explains why licensees appear to be having difficulty meeting a 0.975 EDG target reliability. Decreased EDG reliabilities and/or increased MOOS unavailabilities erode the risk benefits obtained from implementing the SBO rule.

- The operating experience indicates that the SBO rule has increased defense-in-depth. The SBO related hardware and procedures have been used in response to unplanned events and provided additional protection. The SBO rule provides additional defense-in-depth to compensate for potential degradation of the ac offsite power system that may result from deregulation of the electric power industry or longer than expected recovery of offsite power after extremely severe weather conditions.

- A comparison of the value-impact expectations to the outcomes indicates that the value-impact was within the expected range of reductions in public dose-per-dollar of cost. As expected, there was wide variation in plant-specific values and impacts because the SBO rule allowed flexibility. Not expected was the addition of costly power supplies, which accounted for 75 percent of the estimated industry cost impact and explains why the NRC value-impact analysis underestimated the cost by a factor of 4. However, it appears licensees justified the cost of the power supplies by counting on offsetting monetary benefits, such as more operating flexibility from increased EDG allowed outage times. Thus the value was also underestimated. The remaining 25 percent of the estimated industry cost impact appears reasonable, considering the outcomes: known coping capabilities, industry risk reduction from plant-specific procedural and hardware enhancements, and additional defense-in-depth.

A comparison of the SBO rule expectations to the corresponding outcomes indicates that resolution of the generic issue of SBO was effective as no additional generic actions are warranted and no new generic safety issues have been identified.

Although the SBO rule was effective for the reasons stated above, consistent with adhering to Principles of Good Regulation that include clarity (coherent and practical regulations) and reliability (regulations based on operating experience) there are opportunities to revise the regulatory guidance and inspection documents. The proposed revisions are not intended to impose any new regulatory requirements, are consistent with the SBO technical basis (NUREG–1032); ensure high levels of EDG reliability; maintain present levels of safety by ensuring the risk benefits obtained from implementing the SBO rule do not erode; provide practical guidance for reactor shutdowns with limited offsite or onsite power sources; and use operating experience to improve the predictability and consistency of NRC decisions in the area of EDG reliability. The opportunities are as follows:

(1) Regulatory Guides 1.155, “Station Blackout,” August 1988; RG 1.9, “Selection, Design, Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants,” Rev. 3, July 1993; and RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2, March 1997; which address use of the existing EDG reliability terms, criteria, and measurements may need to be revised in a coherent manner to: (a) clarify that EDG unavailability due to MOOS with the reactor at power should be included in the reliability calculation; (b) clarify that licensees should balance increased EDG reliability against the increased EDG unavailability to maintain the RG 1.155 minimum individual EDG target reliabilities; (c) clarify that the EDG system boundary used in the reliability calculation should include the load sequencer and the bus between the EDG and the loads; and (d) establish common EDG start and load-run criteria for the guidance.

Inspection documents Temporary Instruction 2515/125, ?Inspection of Implementation of Station Blackout Rule,” (no date); and Inspection Procedure 62706, “Maintenance Rule,” December 31, 1997; may need revision to delete use of the Nuclear Management and Resources Council (NUMARC)(now NEI) trigger values to assess compliance with the 0.95 and 0.975 EDG target reliability. The NUMARC trigger values, which are not endorsed by the regulatory guidance, do not provide high levels of confidence that the EDG target reliability is being met; this is inconsistent with ensuring high EDG reliability, delays corrective action, and erodes the risk benefits obtained from implementing the SBO rule.

(2) Operating events indicate that the availability of some Aac power supplies is dependent on offsite or onsite power supplies. SBO-related inspection documents may need inspection attributes to verify that the Aac sources meet NUMARC 87-00, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,” Revision 1, May 1993; Appendix B, B.8, “Minimal Potential For Common Cause Failure.”

(3) Regulatory Guide 1.93, “Availability of Electric Power Sources,” December 1974, the basis for technical specifications in the area of ac onsite and offsite power supply availability, provides for shutdown of the reactor following extended ac power supply unavailability. Plant shutdown with one or more offsite or onsite power supplies unavailable could exacerbate the grid condition or remove redundant sources to operate decay heat removal systems, increasing the likelihood of an SBO. Additional practical guidance may minimize the likelihood of an SBO.

(4) Follow-up at 2 plants found a large difference between the unit average EDG reliability based on load-run tests and unplanned demands and the reliability calculated by the licensee based on the last 100 start and load-run tests and unplanned demands. This difference confirms a previous finding in INEL-95/0035, "Emergency Diesel Generator Power System Reliability," February 1996, that the current testing and inspection activities (as prescribed by the NRC) may not be focusing on the dominant contributors to unreliability during actual demands. Accordingly, NRC inspection documents may need to be modified to better factor in the conditions and experiences gained from actual system demands to facilitate inspection of EDG compliance.

As lessons learned: (a) to the extent that the NRC staff revises existing regulatory documents to be more risk-informed and performance-based, they may need to be modified to ensure consistent interpretation and use of terms, goals, criteria, and measurements; and (b) new regulations or the accompanying regulatory documents should include quantitative objectives to facilitate evaluation of its regulatory effectiveness.

1 INTRODUCTION

The Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) is reviewing regulations, starting with the station blackout rule (SBO), to determine if the requirements are achieving the desired outcomes. This initiative is part of an evolving program to address regulatory effectiveness to make NRC activities and decisions more effective, efficient, and realistic.

SBO can be a significant contributor to core damage frequency (CDF) and, with the consideration of containment failure, can be an important contributor to reactor risk. In 1980, the Commission designated the SBO issue as Unresolved Safety Issue (USI) A-44, "Station Blackout," to determine the need for additional safety requirements. On June 21, 1988, the NRC concluded that additional SBO safety requirements were justified and published the SBO rule in the *Federal Register* Notice 23203 (Title 10 of the *Code of Federal Regulations* Section 50.63 [10 CFR 50.63], "Loss of all alternating current power") [Ref. 1]. The amendment was intended to provide further assurance that a loss of both offsite and onsite emergency ac power systems would not adversely affect public health and safety.

In May 1997, the staff briefed the Commission on the individual plant examination (IPE) insight report, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," NUREG–1560, December 1997 [Ref. 2]. The report concluded that SBO remains a dominant contributor to the risk of core melt at some plants, even after implementation of the SBO rule; that SBO frequencies are above 1E-05 per reactor year (RY) for many plants; and that these results warrant further investigation of strategies for reducing SBO frequencies. In a staff requirements memorandum, "Briefing on IPE Insight Report," May 28, 1997 [Ref. 3]; the Commission asked the staff to provide a scope and schedule for using IPE results to assess regulatory effectiveness in resolving major safety issues. The staff responded in SECY–97-180, "Response to Staff Requirements Memorandum of May 28, 1997, Concerning Briefing on IPE Insight Report," August 6, 1997 [Ref. 4], noting that the probabilistic risk assessment (PRA) Implementation Plan was tracking activities to assess the regulatory effectiveness of major efforts to resolve safety issues, including SBO. SECY–97-180 noted that these activities would determine whether additional generic action is warranted, assess whether any new generic safety issues (GSIs) warrant attention, and whether the IPE results justify plant-specific actions.

2 BACKGROUND

2.1 Station Blackout and the Station Blackout Rule

Federal Register Notice 23203 amended the regulations to define an SBO and add the SBO rule. In 10 CFR 50.2, "Definitions," SBO is defined as the complete loss of ac electric power to the essential and nonessential electric switchgear buses in a nuclear plant (i.e., loss of the offsite electric power system concurrent with a turbine trip and unavailability of the emergency ac power system). In 10 CFR 50.63, the SBO rule requires that nuclear power plants be capable of withstanding an SBO for a specified duration and of maintaining core cooling during that period. The specified duration would be determined for each plant by comparing the individual plant design with factors that have been identified in NRC technical studies as the

main contributors to the risk of core melt resulting from an SBO. These risk factors are identified in the SBO rule as (1) the redundancy of the onsite emergency ac power sources, (2) the reliability of the onsite emergency ac power sources, (3) the frequency of loss of offsite power (LOOP), and (4) the probable time needed to restore offsite power.

The SBO rule requires licensees to propose and justify an SBO coping duration based on their ability to: (1) maintain highly reliable onsite emergency ac electric power supplies; (2) ensure that the plants can cope with an SBO for some period of time based on the probability of an SBO at the site and the capability to restore power to the site; (3) develop procedures and training to restore offsite and onsite emergency ac power should either become unavailable; and (4) if necessary, make modifications necessary to meet the SBO rule requirements. The SBO rule also requires that the staff do a regulatory assessment of each of the licensee's response to the SBO rule and notify the licensee of the staff's conclusions.

Before the SBO rule, some regulatory documents addressed SBO risk factors such as the unavailability of either offsite or onsite power sources and emergency diesel generator (EDG) reliability. Regulatory Guide (RG) 1.93, "Availability of Electric Power Sources," December 1974 [Ref. 5], provides limiting conditions for reactor unit operation if offsite or onsite power sources are unavailable. RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977 [Ref. 6], expected licensees to periodically demonstrate EDG reliability by means of 69 consecutive valid tests per plant or 23 per EDG with no failures. RG 1.108 discussed that Branch Technical Position EICSB 2, "Diesel-Generator Reliability Qualification Testing," of the Standard Review Plan established an EDG reliability goal of 0.99 (at 50 percent confidence). RG 1.9 "Selection, Design, Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants," Rev. 2, December 1979 [Ref. 7]; calls for 300 valid start and load qualification tests with no more than three failures.

Before SBO rule was implemented, a survey was documented in NUREG/CR–4557, "A Review of Issues Related to Improving Nuclear Power Plant Diesel Generator Reliability," March 1986 [Ref. 8]; that found that only 7 of 56 nuclear power plant sites had an EDG reliability program; 11 of the 56 sites kept EDG records of the demands and failures experienced by each EDG following RG 1.108; and 3 of 56 sites kept a report of each EDG's reliability. In addition, NUREG/CR–4557 reveals that 11 of 78 nuclear power plants had a unit average EDG reliability significantly below 0.95, and 2 of the 78 plants had a unit average EDG reliability below 0.90.

Appendix A, "Summary of the SBO Technical Bases and Additional Background," is summarized in the next section and gives details about the SBO regulatory documents and terms mentioned in this report.

2.2 Technical Basis for SBO Regulatory Requirements and Guidance

The SBO rule was based on several plant-specific probabilistic safety studies; operating experience; and reliability, accident sequence, and consequence analyses completed between 1975 and 1988. In 1975, WASH-1400, "Reactor Safety Study," 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 SBO issue as USI A-44 and the staff completed several technical studies to determine if any additional safety requirements were needed. NUREG–1032, "Evaluation of Station Blackout at Nuclear Power Plants," June 1988 [Ref. 9], 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 Regulatory Guide 1.155, "Station Blackout," August 1988 [Ref. 10]. NUREG–1032 provided the bases for RG 1.155 by analyzing the effect of variations in various offsite and onsite ac power system designs and plant locations, EDG reliability, and SBO coping capability on the SBO CDF. RG 1.155 allows flexibility for implementing the SBO regulatory requirements plant-specific basis. Consequently, the values and impacts associated with the SBO rule were expected to vary significantly. NUREG–1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, 'Station Blackout,'" June 1988 [Ref. 11], established risk reduction expectations, estimated industry and NRC implementation costs, and used these factors to analyze the value-impact. The public and the industry provided comments on the drafts of the SBO Rule, RG 1.155, and NUREG–1109. The industry comments resulted in significant increases in the estimates of industry costs to implement the SBO rule.

As part of the industry involvement in developing the SBO rule, the Nuclear Management and Resources Council (NUMARC) (now NEI) submitted NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," November 1987, [Ref. 12] as an alternative to complying with the SBO rule. In content, NUMARC 87-00 followed RG 1.155 and added prescriptive, practical guidance which was endorsed by RG 1.155.

The NRC SBO initiatives expected that the resolution of GSI 56, "Diesel Generator Reliability," would elaborate 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. 13] recommended that the staff incorporate (by reference or example) EDG unavailability (i.e., maintenance [and testing] out of service [MOOS]) 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. 14]) into the regulatory guide and the NUMARC guideline being developed for the maintenance rule. The recommendation was implemented in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," June 1993, [Ref. 15] and NUMARC 93-01, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plant," May 1993 [Ref. 16].

The NRC completed safety evaluations for each plant (they can be found in the NRC Public Document Room under each plant's docket number). The NRC also completed eight pilot inspections by October 1994 using Temporary Instruction 2515/120, "Inspection of Implementation of Station Blackout Rule," (no date) [Ref. 17]. Inspection Procedure (IP) 62706, "Maintenance Rule," December 31, 1997 [Ref. 18], concerns SBO-related EDG performance verifications under the maintenance rule.

3 ASSESSMENT OF THE STATION BLACKOUT RULE

The scope of the SBO rule assessment is to determine whether the SBO rule is effective and whether certain areas may need the staff's attention. This assessment focuses on the SBO

rule and related SBO regulatory documents, and not plant specific issues. The assessment reviews and uses plant specific risk and reliability information to make conclusions about the adequacy of the regulatory documents, the assessment does not address plant-specific issues as these continue to be identified and addressed elsewhere in the regulatory process.

3.1 Method for Assessing Regulatory Effectiveness of the Station Blackout Rule

For the purposes of this assessment, the regulatory documents present expectations (desired outcomes) in terms of specific objectives, requirements, and guidance. Hence, the requirements are an integral part of the set of "expectations" which are collectively used as the basis for conducting effectiveness review.

The regulatory documents are considered effective if the expectations are being achieved. Discrepancies between expectations and outcomes prompted a review of the related regulatory documents to find areas that may need attention. The expectations were established from objective measures stated in the SBO rule, RG 1.155, 53 FR 23203, and NUREG–1109 in the areas of coping capability, risk reduction, EDG reliability, and value-impact. The use of multiple regulatory documents provided an examination of the regulatory process. The value-impact assessment was used to determine if the NRC costs and industry costs to implement the SBO rule were reasonable.

Plant-specific data on the actual SBO rule outcomes relative to the expectations for SBO coping capability, selected EDG target reliability, and modifications are shown in Appendix B, "Plant-Specific SBO Information by Reactor Type and Operating Status." The data were collected only from publicly available sources such as licensee PRA/IPEs dated from November 30, 1991 to July 27, 1994, as recorded in the NRC PRA/IPE databases, licensee event reports (LERs), and NRC/licensee correspondence, particularly that related to the SBO rule safety evaluations. When using the above PRA/IPEs it should be recognized that their data are estimates do not always reflect the current design or operating performance of safety systems.

Completed NRC system risk and reliability studies that analyzed the dominant SBO risk factors were also a source of data to measure outcomes. INEL-95/0035, "Emergency Diesel Generator Power System Reliability," February 1996 [Ref. 19], presents an analysis of the reliability of some of EDG power systems at U.S. nuclear plants during the period 1987–1993 and includes SBO insights relative to risk and EDG reliability. INEL-95/0035 is being updated to reflect the operating experience through 1998 for 100 percent of the industry and is presently going through internal comment process as "Reliability Study Update: Emergency Diesel Generator Power System, 1987–1998 (DRAFT)," December 1999 [Ref. 20], (INEL-95/0035 DRAFT update). Both Idaho National Engineering and Environmental Laboratory (formerly INEL) (INEEL) reports were used. In NUREG/CR–5496, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1996," November 1998 [Ref. 21], the NRC contractor has analyzed LOOP events at U.S. nuclear plants from 1980–1996 to determine their likelihood and duration. Appendix B also contains a tabulation of the NUREG–1032 and NUREG/CR–5496 data on plant-specific LOOP events and notes LOOP events that had recovery times 4 hours or longer.

3.2 Comparison of Expectations and Outcomes

Table 1, “Summary of Station Blackout Rule Expectations and Outcomes,” summarizes the SBO rule (and related regulatory guidance and industry guidelines) in the areas of coping capability, risk reduction, EDG reliability, and value-impact.

Table 1 Summary of Station Blackout Rule Expectations and Outcomes

Station Blackout Rule Expectations		**Actual Outcomes**	**Observations**
Area	**Expected Result**		
Minimum Acceptable Coping Capability	100 plants will analyze and select a 2-, 4-, 8-, and 16-hour coping capability. 100 plants will develop procedures and training. 39 plants complete modifications.	All plants have 4- or 8-hour coping capability. 108 plants developed procedures, completed training. 72 plants completed modifications.	Expectations exceeded.
Industry-Wide Risk Reduction in Mean SBO CDF (delta)	2.6E-05/RY delta	3.2E-05/RY delta	Expectations exceeded. Overall risk reduced. Most vulnerable plants completed initiatives to attain low SBO CDF. RG 1.93 actions to shut down with unavailable power supplies not practical.
EDG Reliability	RG 1.155 plant individual target EDG reliability of 0.95 or 0.975, assuming EDG unavailability due to MOOS while the reactor is in a power or non-power status is negligible.	Overall industry unit average unit EDG reliability is generally better than 0.95 however, the EDG 0.975 EDG reliability target appears to be difficult to achieve considering MOOS while the reactor is at power. MOOS while the reactor is at power is not negligible SBO regulatory documents contain multiple EDG performance bases and expectations.	Individual EDG reliabilities not publically available; used unit average EDG reliability- based safety system demands. Observed EDG performance generally exceeds PRA/IPE EDG performance assumptions; additional SBO CDF reductions possible. However, MOOS and failure to meet EDG target reliability could erode risk benefits from the SBO rule. There are opportunities to clarify the regulatory guidance
Value-Impact	2400 person-rem averted per $M. Expected range was 700 to 5000 person-rem/$M. Based on averting 145K person-rem at a total cost of $61.5M.	954 person-rem averted per $M. Based on averting 178,000 person-rem at a total cost of $187M.	Outcome was near the low end of the expected range. The total costs were underestimated by a factor of 3 due to additions of power supplies.

3.2.1 Risk Reduction

In NUREG–1109, the staff estimated that on an industry wide basis the implementation of the SBO rule would result in an industry risk reduction of 2.6E-05 per RY. This expectation is based on a mean SBO CDF before and after the SBO rule of 4.2E-05 per RY and 1.6E-05 per RY, respectively. In addition, as indicated in NUREG–1109, the staff expected that more plants would be in lower SBO CDF ranges after implementation of the SBO rule. The expected range changes are shown in Table 2, "The Number of Plant Units in Station Blackout Core Damage Frequency Ranges Before and After Station Blackout Rule Implementation" which compares the number of plants in each SBO CDF range before and expected after SBO rule implementation. These data were obtained from NUREG–1109, which estimated a plant's SBO CDF before SBO rule implementation from plant-specific dominant risk factor characteristics identified in NUREG–1032 assuming that all plants could cope with an SBO for at least 2 hours, various plant-specific NRC and licensee PRA/IPEs, and other data available around the 1985 time-frame. Information from Appendix B was used in Table 2 to show the actual outcome as changes in the SBO CDF range numbers after implementation of the SBO rule.

Table 2 The Number of Plant Units in Station Blackout Core Damage Frequency Ranges Before and After Station Blackout Rule Implementation

Parameter	Number of Plants in SBO CDF Range (E-05 per reactor-year)											
SBO CDF Range	< 0.5	0.5 .99	1.0 1.49	1.5 1.99	2.0 2.49	2.5 2.99	3.0 3.49	3.5 3.99	4.0 4.49	4.5 4.99	5.0 9.99	10 35
Before SBO rule Implementation (Estimated)	5	13	14	7	13	4	9	5	4	3	13	10
Expected After SBO rule Implementation	23	23	14	9	6	5	6	5	4	0	5	0
Actual Outcome After SBO rule Implementation	46	22	13	17	1	3	1	3	0	1	1	0

On the basis of the SBO CDF data for all plants in Appendix B, the mean SBO CDF associated with SBO rule implementation is 1.0E-05 per RY. Therefore, the reduction in the estimated mean SBO CDF was approximately 3.2E-05 per RY, slightly better than the 2.6E-05 per RY expected. Comparison of the range numbers of the expected to the actual outcome after SBO implementation in Table 2 shows more plants have lower SBO CDFs and fewer plants have higher CDFs than expected. The actual outcome is even more favorable when compared to the corresponding SBO CDF range values before SBO rule implementation.

Appendix C, "Comparison of Selected SBO Characteristics," was prepared from the data in Appendix B to facilitate comparisons and analyses of SBO characteristics. The comparison of the actual LOOP initiating frequencies while at power from 1968 to 1996 (obtained from NUREG–1032 and NUREG/CR–5496) to the values used in the PRA/IPEs shows that the PRA/IPE LOOP initiating frequencies may have been underestimated by more than a factor of 2–39 for eight plants. Analysis of Appendix C also indicates that the SBO rule was effective in

addressing the plants most vulnerable to extremely severe weather and plant-centered LOOPs. In addition, the 15 of the 21 plants that have the greatest vulnerability to a plant-centered LOOP (a LOOP initiating frequency greater than 1.0E-01 per RY) have access to an alternate ac (Aac) power supply, and 19 of the 21 plants have insignificant SBO CDFs with an order of magnitude of 1.0E-06 or smaller. Last, the 8 operating plants that have the highest expected frequency of extremely severe weather frequency (Category 5) and the 10 plants that have an 8-hour coping time have an Aac power supply.

Table 3, "Probabilistic Risk Assessment/Individual Plant Examination Sensitivity Analyses," was prepared from information in plant-specific PRA/IPE sensitivity analyses to show the effect of typical SBO rule modifications on the overall risk. Licensees modified PRA/IPEs based on plant-specific hardware and procedural modifications related to the SBO rule implementation. The modified PRA/IPEs consistently showed a reduction in the plant CDF, thus providing strong evidence for the risk reduction gained from promulgation of the SBO rule. Table 3 lists the modifications due to the SBO rule addressed in the PRA/IPE sensitivity analysis and the corresponding risk reduction associated with the modification as a percentage of the total plant CDF. Table 3 indicates that the overall risk (i.e., the risk from SBO and other initiators) dropped for plants that made major SBO rule modifications, such as adding a power supply, or even simple SBO rule procedural changes (e.g., shedding dc loads).

Table 3 Probabilistic Risk Assessment/ Individual Plant Examination Sensitivity Analyses

Description of Modification	Effect on Overall Risk (Percent Reduction of Plant CDF)
Adding EDGs Calvert Cliffs (one safety and one nonsafety EDG) Turkey Point (two safety EDGs)	 24 20
Adding safety EDG Diablo Canyon	 14–18
Add nonsafety EDG for site Arkansas Nuclear 1 Arkansas Nuclear 2	 23–36 43–47
Procedural Arkansas Nuclear 1: EDG service water supply valve open Monticello: Depressurize during SBO Monticello: Battery load shed	 7 17 17
Credit of combustion turbine generator Fermi	 10
Extend battery life from 2 to 4 hours Arkansas Nuclear 1	 16
Improve reliability of onsite gas turbine generator Point Beach	 13

Previous Assessment of the Effect of the Station Blackout Rule on Core Damage Frequencies

The NRC staff evaluated the impact of the SBO rule on CDFs in NUREG–1560. NUREG–1560 concluded that SBO remained a dominant contributor to risk at many plants, even after implementation of the SBO rule, SBO frequencies exceed 1E-05 per RY for many plants, and that the results warrant further investigation of strategies for reducing SBO frequencies. NUREG–1560 provides SBO risk perspectives based on comparing the SBO CDFs for the 56 plants whose PRA/IPEs addressed the impacts of the SBO rule to the SBO CDFs for 51 plants where the impact of the SBO rule on SBO CDF was not known or credited, or was assumed in the PRA/IPEs to have no impact. NUREG–1560 observed that (1) the average reported reduction in total CDF is consistent with the average reduction in SBO CDF from the backfit analysis of the SBO rule; (2) the average SBO CDF for all plant units considered in the evaluation is comparable to a "typical" estimate in an evaluation of SBO accidents at nuclear plants; and (3) the large variability in the SBO CDF results for the plant units evaluated is also consistent with the variability in the SBO CDF results from other SBO studies.

Generic Safety Issue 23 "Reactor Coolant Pump Seal Failure"

In the memorandum "Closeout of Generic Safety Issue 23, Reactor Coolant Pump Seal Failure, November 8, 1999, the staff also reviewed SBO coping analyses using reactor coolant pump (RCP) seal leakage rates from their research. The staff concluded that no additional cost-beneficial generic requirements should be proposed in part because implementation of the SBO rule has added alternate power sources and reduced the likelihood of an RCP seal LOCA.

Plant Shutdown With Power Supply Unavailability May Increase the Likelihood of an SBO

RG 1.93, "Availability of Electric Power Sources," December 1974, [Ref. 22] provides for the nuclear unit to shut down following extended unavailability of either the offsite power source or emergency onsite power sources, and is the basis for technical specifications in this area. Plant shutdown with the one or more offsite or onsite power supplies unavailable could exacerbate the grid condition or remove redundant sources to operate decay heat removal systems. The extended unavailability of one or more offsite or onsite power supplies should prompt an alternate approach, such as assuring the immediate availability of coping systems, reducing power, or assuring availability of adequate electric grid reserves.

Assessment

The SBO rule was effective in achieving the desired reduction in the SBO CDF. The reduction in the estimated mean SBO CDF was approximately 3.2E-05 per RY, slightly better than the 2.6E-05 per RY expected. The SBO rule caused meaningful reductions in the risk at many plants. Also, more plants have lower CDFs and fewer plants have higher CDFs than expected. As a result of the SBO rule, the plants with the most LOOPs from plant events and extremely severe weather improved the most. Consequently, these plants have relatively low SBO CDFs. However, the mean LOOP initiating frequencies of eight plants used in the PRA/IPE analysis may have been underestimated by factors of 2–39 in comparison to the actual number of plant LOOPs experienced. In addition, RG 1.93, which addresses plant shutdown following extended unavailability of offsite and onsite power supplies, potentially increases the likelihood of an SBO.

3.2.2 Emergency Diesel Generator Reliability

Emergency Diesel Generator Performance

After implementation of the SBO rule, all licensees committed to establish an EDG reliability program and to maintain a minimum individual EDG target reliability of 0.95 or 0.975. Accordingly, all licensees have committed to an information and data collection system to monitor the achieved reliability levels and compare them to the target values. In addition, all licensees have monitored EDG unavailability since 1989.

INEL-95/0035 and the INEL-95/0035 DRAFT update investigated the performance of EDG trains on demand. The INEL-95/0035 DRAFT update was used in this assessment even though it contains preliminary results. However, the INEL-95/0035 DRAFT, has more data on the time period since the implementation of SBO rule and better illustrates the EDG reliability outcome of SBO implementation.

INEL-95/0035 and the INEL-95/0035 DRAFT compare the EDG train performance on demand to the RG 1.155 EDG target reliability expectations. These comparisons are somewhat unequal. The INEEL studies compare the unit average EDG reliability based on performance of its “safety mission” (a loss of voltage to the safety buses that results in both a start and a load-run) to the RG 1.155 0.95 and 0.975 individual EDG reliabilities. Licensees generally calculated RG 1.155 individual EDG reliability using data from plant-specific valid starts and load-runs over different time interval. The INEEL studies calculated a unit average EDG reliability using (a) 12 years of test and unplanned demand data that simulate the safety mission and (b) Bayes’ empirical methods, which allow consideration of the plant-specific and industry data to establish new estimates of the mean reliability and its uncertainty bounds. Nevertheless, the INEEL comparison is valid. Since the 0.95 or 0.975 individual EDG reliability is the target, demonstrations of the EDG’s ability to perform their safety mission over several years should result in unit average EDG reliabilities of 0.95 or 0.975 or higher; lower would indicate the individual EDG reliabilities are not consistent with RG 1.155.

The INEEL reports evaluated the validity of the RG 1.155 (Section B) assumption that so long as the EDG unavailability due to MOOS is not excessive, the maximum specified EDG reliabilities of 0.95 and 0.975 would result in an acceptable overall reliability for the emergency power system. INEEL evaluated only the EDG MOOS with the reactor at power as it has higher risk due to the possibility of a demand while the EDG is unavailable and the reactor is at power. Appendix A explains MOOS in detail.

The following insights were obtained from the INEEL analyses of EDG safety performance:

(1) The INEL-95/0035 DRAFT update indicates that the 58 plants that committed to a 0.95 minimum EDG target reliability achieved a unit average EDG reliability of 0.945 to 0.989, with and without consideration of MOOS with the reactor at power; INEL-95/0035 EDG reliabilities are similar ranging from 0.934 to 0.99.

(2) INEEL studies indicate that the plants that committed to a 0.975 minimum individual EDG target reliability are having difficulty achieving a 0.975 unit average EDG target reliability. With MOOS while the reactor unit is at power, the INEL-95/0035 DRAFT update indicates

only 8 of the 44 operating plants that committed to a 0.975 minimum EDG reliability achieved a unit average EDG reliability above 0.975; plant reliabilities ranged from 0.917 to 0.98. INEL 95/0035 EDG reliabilities are similar: none of the 19 plants considered achieved a unit average EDG reliability of 0.975. Without MOOS, the INEL-95/0035 DRAFT update indicates that approximately 36 of the 44 plants that committed to a 0.975 minimum EDG reliability had a 0.975 or higher unit average EDG train reliability; INEL 95/0035 differs: 18 of the 19 plants considered had a 0.975 or better unit average EDG reliability.

Under RG 1.155 licensees, could select the 0.975 EDG target reliability to achieve shorter coping durations to avoid modifications. In addition, the plants with the least independence and redundancy in onsite emergency power supplies for safe shutdown equipment had to select the 0.975 minimum EDG target reliability to achieve a lower SBO CDF. Consequently, failure to meet the 0.975 EDG target reliability could significantly erode these risk reductions. The effect of a 0.025 decrease in EDG reliability on the SBO CDF was evaluated using Table 4, “Estimated Increase in The SBO CDF Due to a Decrease in EDG Reliability From 0.975 to 0.95." Table 4 was developed from NUREG-1032, Table C.4, “Tabulated estimated values of total core damage frequency for SBO accidents as a function of EDG configuration, EDG reliability, offsite power cluster, and ability to cope” assuming the coping duration was constant. NUREG–1032 grouped each plant’s offsite power system into one of five offsite power clusters based on its susceptibility to grid and weather conditions. Table 4 illustrates that a 0.025 decrease in EDG reliability could increase the SBO CDF by 1.0 E-05 per RY or more in plants in the offsite power clusters 2–5 (about 60 plants). Increases in the SBO CDF of 1.0 E-05 per RY or more erode the 3.2 E-05 per RY risk reduction obtained from implementing the SBO rule.

Table 4 Estimated Increase in the Station Blackout Core Damage Frequency Due to a Decrease in Emergency Diesel Generator Reliability From 0.975 to 0.95

Emergency ac power system configuration	Coping duration	Offsite power cluster				
		1	2	3	4	5
1/2	4 8	<E-05 <E-05	1.1E-05 <E-05	2.6E-05 <E-05	6.6E-05 1.7E-05	1.4E-04 5.1E-05
2/3	4 8	<E-05 <E-05	3.2E-05 1.1E-05	8.0E-05 2.0E-05	0.3E-04 4.5E-05	4.5E-04 1.3E-04
2/4	4 8	<E-05 <E-05	<E-05 <E-05	<E-05 <E-05	1.3E-05 <E-05	4.8E-05 <E-05

(3) The INEEL studies indicate that MOOS has a significant effect on the reliability calculation. Table 5, “Effects of MOOS While the Reactor is at Power on EDG Reliability,” was prepared from data in the INEL-95/0035 DRAFT update. Table 5 indicates that consideration of MOOS while the reactor is at power lowers the unit average EDG train

reliabilities for the 0.95 and 0.975 groups by 0.034 and 0.012, respectively. INEL-95/0035 gives similar results.

Table 5 Effects of Maintenance Out of Service
While the Reactor is at Power on Emergency Diesel Generator Reliability

EDG target reliability	Mean industry unit average EDG reliability		
	without MOOS	with MOOS	decrease in reliability
0.95	0.985	0.954	0.034
0.975	0.978	0.967	0.012

NUREG/CR–5994, “Emergency Diesel Generator: Maintenance and Failure Unavailability, and Their Risk Impacts,” October 1994, [Ref. 23] analyzed the effects of MOOS increases on the CDFs for five plants. The data indicate that when MOOS increases from 0.007 (assumed in RG 1.155 Section B) to 0.02, the plant CDF may increase by less than 1.0E-06, which is insignificant; however, an increase in MOOS to 0.04 may cause increases in plant CDF of more than 1.0E-05 and could offset the risk reduction obtained by implementing of the SBO rule.

(4) The INEL-95/0035 DRAFT update indicates that three of 102 operating plants have a unit average EDG reliability less than 0.95 without MOOS. Follow-up at 2 of the 3 plants found that the unit average EDG reliability was 0.99 or higher based on the last 100 EDG start and load-run tests and/or unplanned demands; this is 0.05–0.07 more than corresponding INEEL plant-specific values, which are based only on load-run tests and unplanned demands that simulate the EDG safety mission. This finding may indicate that plant-start testing is not identifying the causes of EDG unreliability during demands that simulate the EDG safety mission. This confirms an INEL 95/0035 finding based on a review of LERs, that the current testing and inspection activities may be missing the dominant contributors to unreliability during actual demands and may need to be modified to better consider the conditions and experiences gained from actual system demands.

(5) The INEEL-95/0035 DRAFT indicates that many licensees could demonstrate additional reductions in the plant SBO CDF from improved EDG performance. In addition, maintaining high EDG reliability levels provides assurance that the PRA/IPE EDG performance assumptions are valid. Each INEEL study has a figure similar to Figure 1, “Preliminary Plant-Specific EDG Unit Average EDG Unreliability Compared to the Unreliability Estimates Using the PRA/IPE EDG Performance Assumptions,” which shows EDG unreliability for 44 plants grouped by 24, 8, and 6 hour mission time. Figure 1 indicates (a) that 0.05 and 0.025 unreliabilities (which correspond to the 0.95 and 0.975), RG 1.155 EDG reliability targets are generally in the upper end of the PRA/IPE EDG uncertainty interval and (b) that except for a few cases, the mean PRA/IPE EDG unreliability values are higher than the corresponding values calculated from the operating experience.

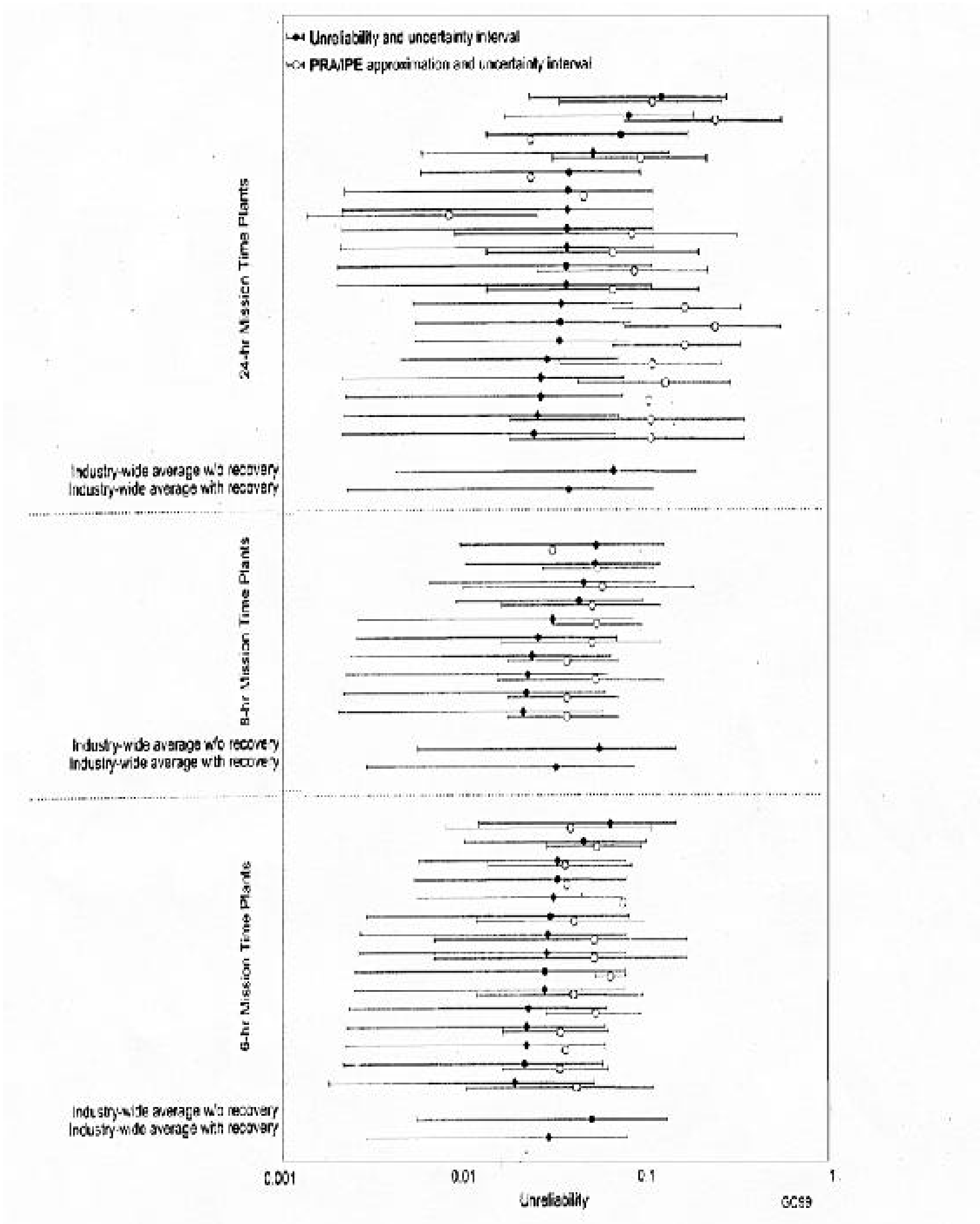

Figure 1 "Preliminary unit average emergency diesel generator unreliability compared to the unreliability estimates using the probabilistic risk assessment/individual plant examination emergency diesel generator performance assumptions."

Summary of EDG Performance Bases

A review of EDG performance bases found that the SBO rule, the resolution of GSI-56 in conjunction with the SBO rule and the maintenance rule, and inspection procedures, resulted in multiple EDG performance bases. These are summarized below. Appendix A provides background information on the regulatory documents discussed below.

1) RG 1.155, Section C.1.2 expects licensees to establish a reliability program to maintain a minimum individual EDG target reliability of 0.95 or 0.975. RG 1.155, Section B assumes that as long as MOOS (regardless of the reactor power status) is not excessive, the maximum EDG failure rates for each EDG could result in an acceptable overall reliability for the emergency power system configuration. RG 1.155, Section C.1.1 expects EDG reliability to be determined using the NSAC-108 definition of reliability as the product of the starts and load-run reliabilities excluding MOOS, and the NSAC-108 definitions of valid start and load-runs. RG 1.155, Section C endorses NUMARC 87-00, November 1987, as acceptable alternative guidance. NUMARC 87-00, Section 6.0 also requires the monitoring of the unit average individual EDG unavailability in comparison to an industry average.

(2) Under RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Rev. 2, March 1997, [Ref. 24] licensees can select the RG 1.155 0.95 or 0.975 individual EDG target reliability as either a goal or a performance criterion (Section B, “Emergency Diesel Generators”); or they can select IPE unavailability values compared to the industry values as a goal or performance criterion (Section B), or maintenance preventable functional failures as the sole performance criterion (Section 1.4). RG 1.160, Section B also discusses the balancing of reliability and unavailability under (a)(3) of the maintenance rule. In addition, RG 1.160 Rev. 2 endorses NUMARC 93-01, “Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plant,” April 1996, [Ref. 25] that allows licensees to use PRA/IPE numerical assumptions about EDG reliability performance as goals or performance criteria under the maintenance rule (NUMARC 93-01, paragraph 9.3.2.).

(3) RG 1.9, Rev. 3 provides that onsite emergency power supplies be selected with sufficient capacity, be qualified, and have the necessary reliability and availability for SBO and design basis accidents. RG 1.9, Rev. 3, Section 2.2 gives definitions of valid EDG starts and load-runs and delineates the ac onsite emergency power equipment and system boundary used to count failures.

(4) SBO rule and maintenance rule inspection documents provide EDG performance bases.

Inspection documents TI 2515/120, Section 2515/120-040 and IP 62706, Section 3.05, December 31, 1997, use the NUMARC 87-00, Rev. 1, Appendix D trigger values (NUMARC trigger values) for assessing compliance with the RG 1.155 minimum individual EDG target reliabilities of 0.95 and 0.975. Appendix D “EDG Trigger Values,” provides background information.

EDG Performance Bases Not Always Clear

Since licensees appear to be having difficulty maintaining the 0.975 minimum target reliability and since there are multiple EDG performance bases, the SBO related regulatory documents on EDG reliability were compared to the SBO technical basis (NUREG–1032), and to each other, to assure they could be readily understood and easily applied to achieving high EDG reliability. The comparison found inconsistent use of EDG performance terms, criteria, and measurements:

(1) The NUREG–1032 EDG equipment and system boundary used to count failures included the load sequencer and the bus between the EDG and the loads. RG 1.155 does not specify the EDG boundary. The RG 1.9 Rev. 3, Section 2.2, EDG system boundary does not include the load sequencer and the bus between the EDG and the loads. The INEL-95/0035 and INEL-95/0035 DRAFT update data indicate that load sequencer failures have caused EDGs to fail after unplanned demands; plant calculations of EDG reliability may be inflated if these failures are not considered.

(2) The NUREG–1032 and RG 1.155 criteria for determining valid start and load-run demands to use in the EDG reliability calculations apply only to actual test or unplanned demands. The corresponding criteria in RG 1.9, Rev. 3, Section 2.2 differ, counting “conditional failures” identified in the course of maintenance that could have caused start or load-run failures.

(3) RG 1.160, Rev. 2 guidance gives licensees the option of monitoring EDG performance using different criteria than the RG 1.155 EDG target reliabilities. Under RG 1.160, Rev. 2, licensees may chose PRA/IPE unavailability, PRA/IPE reliability, or maintenance preventable functional failures. Licensees must balance reliability and unavailability, however it is not clear that licensees should balance the improvement in EDG reliability against the resulting increase in EDG unavailability (MOOS); otherwise the balanced reliabilities could be non-conservative with respect to the RG 1.155 minimum individual EDG target reliabilities. In addition, PRA/IPE reliability and unavailability are represented by a range of values, and some of the values are non-conservative with respect to the RG 1.155 minimum individual EDG target reliabilities. As already shown, a 0.025 decrease in EDG reliability or a 0.04 increase in EDG unavailability results in increases in the SBO CDF that offset the risk reductions obtained from implementing the SBO rule.

(4) TI 2515/120, IP 62706, and some licensees are using the NUMARC EDG trigger values for demonstrating achievement of the 0.95 and 0.975 minimum individual target reliability. However, the NRC intended that the NUMARC trigger values not be used for this purpose. The Advisory Committee on Reactor Safeguards (ACRS) convinced the staff that the NUMARC trigger values provided low statistical confidence that high levels of EDG reliability would be achieved. Others expressed the view that allowing EDG reliability performance to degrade to these levels provides high statistical confidence that the EDG target reliability is not being met however, waiting to attain these levels delays potentially needed corrective actions. Statistics similar to those used by these two perspectives are shown In Appendix D. In a letter to the ACRS dated October 29, 1993, [Ref. 26] the Executive Director of Operations informed the ACRS that the staff agreed that conformance of individual EDGs with trigger values cannot be taken, in any statistical fashion, to mean that the EDG has demonstrated achievement of the licensee’s

commitments to 0.95 or 0.975. The letter stated that Note 3 was added to RG 1.160, June 1993 to emphasize this fact. In addition, 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 (which references NUMARC 87-00, Rev. 1, and uses the trigger values 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 be interpreted to mean NUMARC 87-00, Rev. 1, was not endorsed.

However, wording in past revisions of RG 1.160 could lead to the conclusion that the NUMARC reference documents do not apply to the EDGs and may explain how licensees and the NRC inspection documents adopted the NUMARC trigger values. For example, RG 1.160, June 1993, Section C also states that the example in NUMARC 93-01, Section 12.2.4, describes an acceptable method to establish EDG performance criteria and/or goals and subsequently monitor EDG performance. As another example RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Rev.1, January 1995 [Ref. 27], Section B states that the EDGs are required to be handled under (a)(1) and (a)(2) of the maintenance rule as described in NUMARC 93-01.

Assessment

Before implementation of the SBO rule, only 11 of 78 plants surveyed had formal EDG reliability programs; 11 of 78 plants had a unit average EDG reliability less than 0.95; and 2 of 78 plants had a unit average EDG reliability of less that 0.90. Since implementing the SBO rule, all plants have established an EDG reliability program and an improved EDG reliability. Considering actual safety performance and maintenance and test out of service (MOOS) with the reactor at power, only 3 of 102 operating plants have a unit average EDG reliability below 0.95 and above 0.90. However, analysis indicates MOOS with the reactor at power significantly affects the reliability calculation based on safety performance and explains why licensees are having difficulty meeting the 0.975 EDG target reliability. Decreases in reliability and/or increases in MOOS unavailability erode the risk benefits of implementing of the SBO rule.

There are opportunities to revise the RG 1.155, RG 1.9, and RG 1.160 which inconsistently use EDG reliability terms, criteria, and measurements. These RGs may need to be revised to: (a) clarify that MOOS with the reactor at power should be included in the reliability calculation to ensure high levels of EDG reliability (b) clarify that licensees should balance increased EDG reliability against the increased EDG unavailability to maintain the RG 1.155 minimum individual EDG target reliabilities and the risk levels achieved from implementing the SBO rule, (c) clarify that the EDG system boundary used in the reliability calculation should include the load sequencer and the bus between the EDG and the loads to be consistent with the SBO technical basis (NUREG–1032), and (d) establish common EDG start and load-run criteria.

Inspection documents Temporary Instruction 2515/125 (no date) and IP 62706 may need revision to delete use the NUMARC trigger values to assess compliance with the 0.95 and 0.975 EDG target reliability. Statistically the NUMARC trigger values do not provide high levels of confidence that the EDG target reliability is being met; this inconsistent with ensuring high EDG reliability, delays corrective action, and erodes the risk benefits obtained from implementing the SBO rule.

Follow-up at two plants in the 0.975 EDG reliability group that had less than 0.95 unit average EDG reliability based on load-run tests and unplanned demonstrations of its safety mission found

the unit average EDG reliability based on the last 100 valid start and load-run tests and unplanned demands was 0.05–0.07 higher; this confirms a previous finding in INEL-95/0035 that the current testing and inspection activities may be missing the dominant contributors to EDG unreliability during actual demands and may need to be modified to better consider the conditions and experience gained from actual system demands.

3.2.3 Minimum Acceptable Coping Capability, Plant Procedures, Training, and Modifications

In NUREG–1109, the staff expected that 100 plants would (1) be able to show a minimum acceptable coping capability of 2, 4, 8, or 16 hours based on plant-specific characteristics; (2) complete an analysis of the plant's ability to cope with an SBO for the selected duration; (3) develop SBO-related procedures; (4) complete training on these procedures; and (5) complete modifications necessary to cope. The SBO rule was flexible allowing for a wide range of coping capabilities, so plants with an already low risk from SBO could select short coping times and would need few, if any, modifications. Plants with higher risk could select longer coping times and would possibly need modifications to cope. Thirty-nine plants were expected to complete hardware modifications.

Appendix E, "Station Blackout Rule Activity and Modification Summary," was prepared to show the expected number of plants completing analyses, procedure development and training, various types of modifications in the licensee's response to the SBO rule, the estimated costs of the modifications, and the outcomes. The costs are discussed in Section 3.2.4, "Value-Impact Analysis."

The outcome was that 108 plants selected a minimum SBO coping capability of 4 or 8 hours, completed the coping analysis, developed procedures, completed training, and 72 plants completed modifications. Not credited was the fact that some plants may have developed adequate procedures and completed training before the SBO rule in response to Generic Letter 81-04, "Emergency Procedures and Training for Station Blackout Events," February 25, 1981.

<u>Assessment</u>

The SBO rule expectations were met in the areas of coping analysis, procedure development and training, and modifications. The scope and number of modifications to achieve specified coping durations exceeded the expectations and may explain why risk reductions were greater than expected.

3.2.4 Value-Impact Analysis

A comparison of value-impact expectations and outcomes was derived from NUREG–1109. The value-impact analysis in NUREG–1109 estimated the expected value based on the public dose reduction associated with the SBO rule. The impacts were based on estimates of industry and NRC costs to implement the SBO rule. As explained in Appendix A under "Public and Industry Comment," the industry costs were provided by the industry. The ratio of the value to the impact was also derived as an indication of the cost effectiveness of the SBO rule. Each of these matters is discussed below.

Table 6, “Station Blackout Rule Value-Impact Summary,” was prepared from Appendix E to compare the expected impact, value, and value-impact ratio to the corresponding outcomes. Appendix E used the expected values and impacts from NUREG–1109. The outcomes were estimated using either the values and impacts from NUREG–1109 or from information submitted to the NRC by the licensees. The cost of an additional EDG at Davis Besse was $9.07M. This value was taken as a representative amount for the addition of power supplies and used in the cost estimates shown below.

Table 6 Station Blackout Rule Value-Impact Summary

Value Impact Factors	Expected ($)	Outcome ($)
Impact–NRC and Industry Implementation Cost		
Best estimate		
Miscellaneous SBO modifications	60M	63M
NRC implementation	1.5	2.0M
19 additional power supplies	0	174M
TOTAL	61.5M	237M
Monetary savings attributed to adding additional power supplies for operating flexibility		-50M
TOTAL		187M
Estimated range		
Total	43M–95M	–
Plant-specific	350K–4M	350K–20M
Estimated Value		
Public dose reduction in person-rem	145,000	178,000
Value-Impact Ratio (person-rem averted/$million)		
Best estimate	2400	954
Range	700–5000	–

In NUREG–1109, the staff used information supplied by the industry to estimate that the industry would spend approximately $60 million (M). NUREG–1109 estimated that the NRC would spend approximately $1.5M to implement the SBO rule. The total estimated SBO rule implementation cost of $61.5M was a best estimate with a low of $43M and a high of $95M. In NUREG–1109, the staff recognized that there would be wide variation in plant-specific costs, ranging from $0.35M for plants that needed only procedural changes to $4M if all the anticipated modifications were completed. The outcome in terms of actual SBO rule costs was about $237M, which exceeded expectations by about a factor of 4. The discrepancy is attributable to the addition of 19 power supplies at an estimated cost of approximately $174M. These additions, although consistent with the SBO rule, were also motivated by monetary benefits from operating flexibility from these power supplies. So the NRC also underestimated the value and it may be conservative to ascribe all these costs to the SBO rule.

The NRC estimated that the added power supplies added an estimated $50M in value from more operating flexibility from increased allowed outage times (AOTs). For example, Davis-Besse obtained NRC approval to change its technical specifications to increase the AOT for its EDGs from 3 to 7 days to gain flexibility in performing EDG maintenance while the reactor is at power. In submitting this change to the NRC, Davis-Besse noted that its SBO diesel generator installation would save $5.25M over the remaining plant life, including $3.15M for increased flexibility in performing maintenance on its safety EDGs and $2.1M in replacement power costs. North Anna increased its EDG AOT from 3 to 14 days by crediting the non-Class 1E SBO EDGs. Other licensees increased their EDG AOT by documenting excess EDG redundancy or Aac power supply connections from their SBO analyses. RES estimated that 10 reactor units besides Davis-Besse benefitted $50M from increased EDG AOT to gain operating flexibility.

It also appears that the addition of a power supply resulted in significant plant-specific risk reductions. Turkey Point added two EDGs and cross-ties reducing the risk of an SBO after a LOOP from 7.6E-04 to 2.9E-06 per RY. Table 3 indicates that adding power supplies to the Calvert Cliffs, Diablo Canyon, and Arkansas Nuclear One PRA/IPEs resulted in risk reductions ranging from 14 to 47 percent.

In NUREG–1109, the staff expected that the SBO rule would add value by averting 145,000 person-rem from an accident; the estimates ranged from 216,500 person-rem to 65,000 person-rem. The expected reduction in person-rem was calculated by multiplying the reduction in CDF per RY from an SBO by the remaining life of the plant (assumed to be 25 years) and the estimated public dose based on the highest source term at a site from an accident. The total reduction in person-rem for each plant was summed and divided by 10 to consider the smaller source term for an SBO at a 50-mile radius. The 145,000 person-rem derived in NUREG–1109 appears to be realistic, and may be low since the weighted population dose factor for the five NUREG–1150 power reactors ranges from 166,000 to 2,000,000 person-rem within 50 miles of the plant (NUREG/BR–0184, “Regulatory Analysis Technical Evaluation Handbook,” February 1997).

In 1109, the staff calculated a value-impact ratio of 2,400 person-rem per $M based on averting 145,000 person-rem at a cost of $61.5M with a range of 700 person-rem per $M to 5,000 person rem per $M. The outcome was a value-impact ratio of 954 person-rem per $M based on an estimated impact of $187M, which is within the expected range. The outcome recognizes that received approximately $50M in monetary benefit from increased EDG AOT, reducing the impact from $237M to $187M.

Previous Assessment of the Cost Effectiveness of the SBO Rule

The staff also evaluated the industry’s average cost per person-rem averted in satisfying the SBO requirements in SECY–97-180, “Response to Staff Requirements Memorandum of May 28, 1997, Concerning Briefing on IPE Insight Report,” August 6, 1997. The staff used a different methodology and concluded that, on average, the SBO rule averted a person rem at a cost of $4,750 (211 person-rem per $M); and that this cost is likely too high because it does not give full credit for other sizable economic benefits and it is skewed by a few plants whose SBO cost exceeded $10M. Most reactors incurred SBO costs of less than $1M. A supporting calculation suggested that about 70 percent of the reactors incurred costs of less than $1000 per person rem averted (1000 person-rem per $M), and 75 percent incurred costs of less than $2000 per person-rem averted (500 person-rem per $M).

Assessment

Comparing of the expected value-impact to the actual value-impact indicates that the actual value impact was within the expected range of reduction in public dose-per-dollar of cost. However, the NRC staff did not anticipate that so many licensees would install additional safety-related and nonsafety-related power sources which accounted for 75 percent of the estimated industry cost. So the staff underestimated the expected (“best estimate”) cost by a factor of approximately 4. However, it appears licensees justified the cost of the added power supplies based on offsetting monetary benefits, such as more operating flexibility from increased EDG AOT. So the NRC also underestimated the value. Disregarding the costs of the power supplies, the costs of meeting the SBO rule requirements were reasonable since plants have added SBO coping and recovery procedures, and have established EDG reliability programs to maintain EDG target reliability levels. Further, the SBO rule focused NRC and industry resources on an area known to be important to the overall risk of an operating nuclear power plant, and the SBO rule analysis helps to maintain an acceptable level of safety at operating nuclear power plants.

The industry and NRC costs to implement the SBO rule were reasonable, considering the outcomes. As expected there was wide variation in plant-specific values and impacts because the SBO rule provided flexibility; plants with higher risk needed longer coping times and possibly more modifications, and plants with an already low risk from SBO needed shorter times to cope and fewer modifications.

3.2.5 Insights From Operating Experience Reviews

Performance during operating events provides a additional means to assess the effectiveness of regulatory requirements and guidance. The following are some operating experience insights.

Modifications Due to the Station Blackout Rule

Plant modifications and procedures due to the SBO rule have been relied upon to provide protection during operating events. For example, in the original emergency power system design for Turkey Point Units 3 and 4, the two units shared two 2850-kW safety-related EDGs and five onsite nonsafety-related EDGs that could be connected to the reactor unit emergency power system through a nonsafety-related switchgear. As a result of the SBO rule, the licensee added two 3095-kW safety-related EDGs. A March 1993 report, “Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Sation from August 20–30, 1992," indicates that the five onsite nonsafety-related EDGs were unavailable because of water damage to the nonsafety-related switchgear; the total load on the four safety-related EDGs was approximately 3400 kW, and about 3.5 days into the storm, one of the two original 2850-kW EDGs tripped and was restarted in 2.5 hours. Had the two safety-related 3095-kW EDGs not been added as a result of the SBO rule, the load on the only remaining 2850-kW EDG would have exceeded the EDG rating by 19 percent and most likely the EDG would have failed unless unloaded quickly by the operators, leaving no ac power for 2.5 hours.

Offsite Power System Deregulation

As stated in 53 FR 23203, it was expected that the SBO rule would provide additional defense in depth by requiring plants to cope with an SBO for a specified duration. The increment of defense-in-depth is provided by the preventative effect of reducing the likelihood of core damage

by implementing the SBO rule. In addition to increased coping capability, the SBO rule has resulted in plants gaining increased tolerance to an SBO from reduced risk as previously explained, which addresses the mitigative aspect of defense-in-depth. These SBO rule outcomes provide additional defense-in-depth to compensate for increased risks due to potential degradation of the ac offsite power system that may result from deregulation of the electric power industry as explained in SECY–99-129, "Effects Of Electric Power Industry Deregulation on Electric Grid Reliability and Reactor Safety," May 11, 1999, [Ref. 28] a publicly available NRC technical report, "The Effects of Deregulation of the Electric Power Industry on the Nuclear Plant Offsite Power System: An Evaluation, dated June 30, 1999, [Ref. 29] a recent Information Notice 2000-06, "Offsite Power Voltage Inadequacies," March 27, 2000, [Ref. 30] and Information Notice 98-07, "Offsite Power Reliability Challenges From Industry Deregulation," February 27, 1998, [Ref. 31] .

Dominant SBO Risk Factor Trends

The main factors affecting the likelihood of SBO accidents at nuclear plants are (1) EDG reliability and redundancy, (2) the LOOP frequency, and (3) the time to restore offsite power following its loss.

Neither INEL-95/0035 (with data from 1987–1993) nor INEEL DRAFT update (with data from 1987–1998) found any trend in EDG train performance by year.

Table 7, "Dominant Station Blackout Risk Factor Trends — Offsite Power System," was prepared to compare the offsite power trends in NUREG–1032 (based on 1968–1985 operating experience) to the corresponding offsite power values in NUREG/CR–5496 (based on 1980–1996 operating experience). The results show overall reduction in LOOP frequency and duration. An exception is the increasing duration of grid-related LOOP events. This is being addressed by SECY–99-129.

Table 7 Dominant Station Blackout Risk Factor Trends — Offsite Power System

Source of LOOP Data	**NUREG–1032 (1968–1985 LOOP operating experience)**	**NUREG/CR–5496 (1980–1996 LOOP operating experience)**
Plant-related LOOP		
Mean frequency	0.087	0.04
Median time to restore	18 minutes	20 minutes
LOOPs > 4 hours	0	4
Grid-related LOOP		
Mean frequency (occurrence per year)	0.018	0.0019
Median time to restore	36 minutes	140 minutes
LOOPs > 4 hours	1 (due to severe weather)	0
Weather-related LOOP		
Mean frequency	0.009	0.0066
Median time to restore	4.5 hours	1.2–2.4hours
LOOPs > 4 hours	4	5

Appendix F, “Operating Events,” Table F-1, “Losses of Offsite Power Since 1990 Having Recovery Times of 4 Hours or Longer,” summarizes six plant-centered and three weather-related LOOP from 1990 (when the SBO rule was implemented) to 1998. The LOOPs occurred while the reactor unit was running and it took 4 or more hours to recover offsite power. Three of the events closely followed an SBO resulting in a LOOP and the unavailability or technical inoperability of one EDG. Analysis of the plant events found communication and procedural weaknesses in that delayed recovery. These weaknesses were subsequently corrected and recovery from future losses of offsite power should significantly improve. Analysis of weather events shows that it could take up to 4.5 days to recover offsite power in a hurricane, 28 hours after a tornado, and approximately 8 hours after ice accumulation or contamination from salt sprays. Provisions for these types of events are addressed in NUMARC 87-00, Rev. 1 which requires actions for achieving enhanced coping capability under hurricane and tornado conditions. Key features of NUMARC 87-00, Rev. 1, are (1) actions to be taken in the 24-hour period before anticipated arrival of the hurricane and (2) a commitment to be in a safe shutdown condition 2 hours before the hurricane arrives. Another key feature to mitigate hurricanes and tornados was the addition of Aac power supplies at the plants that have the most vulnerability to extremely severe weather conditions.

Potential SBO Alternate ac Power Source Unavailability

Appendix F, Table F-2, “Station Blackout Challenges,” was prepared from a review of the 1990–1998 operating experience. The review found one event (LER 335/98-007) that identified inadequate SBO recovery procedures during a simulator exercise in 1998 that could have complicated recovery from an SBO. Four events (LER 346/98-006, 247/98-007 and two events in Information Notice [IN] 97-21 [Ref. 32]) identified the potential unavailability during an SBO event of an Aac power source (added as a result of the SBO rule). The latter is a concern since the unavailability of the Aac power supply during an SBO could lead to core damage. IN 97-21 described the dependencies of the Aac support system batteries and ac auxiliary power. LER 247/98-007 reported that the SBO Aac power source output circuit breakers were not capable of being closed onto a de-energized bus during an SBO. The licensee reported that the cause of the problem was insufficient comprehensive testing. LER 346/98-006 reported that a tornado resulted in a LOOP and the technical inoperability of one EDG; and offsite power was restored in 28 hours. An NRC analysis of the tornado event noted that a nonessential bus supplies power to the auxiliaries of the Aac, and if not powered, the batteries would deplete in approximately 28 hours. If the EDGs failed to start or run in the first 8 hours of the 28-hour event, the SBO-DG might have been unavailable likely leading to core damage.

Assessment

Implementation of the SBO rule has strengthened additional defense in depth. As demonstrated at Turkey Point, the SBO rule modifications and procedures have been relied upon to mitigate the consequences of, and provide protection during LOOP events. The SBO rule provides defense in depth if deregulation of the electric utility industry or changing offsite power system trends effect the SBO risk.

There is a lower probability of recovering within the SBO coping time for extremely severe weather LOOP events than from other types of LOOPs. The SBO rule has resulted in industry requirements (that have been endorsed by the NRC) that can be taken before extremely severe weather conditions to mitigate the potential consequences of such an event and the addition of

Aac power supplies at the plants that have the most vulnerability to extremely severe weather conditions.

Subsequent to issuing an IN 97-21, an event revealed that the availability of an Aac power supply system was limited by dependencies on offsite or onsite power supplies and that this situation could lead to core melt.

4 CONCLUSIONS

The report's conclusion is that the SBO rule was effective considering that the risk expectations were achieved, and that industry and NRC costs to implement the SBO rule were reasonable. In implementing the SBO rule, some plants made hardware modifications (e.g., the addition of diesel generator or gas turbine generator power supplies); and all plants generally maintained EDG reliability at 0.95 or better, and established SBO coping and recovery procedures. Consequently, the plants have gained SBO coping capability, reduced risk, increased the tolerance to a loss of ac offsite or onsite power, and many plants benefitted economically from the addition of power supplies. To elaborate:

- The reduction in the estimated mean SBO core damage frequency (CDF) was approximately 3.2E-05 per reactor year, slightly better than the 2.6E-05 per reactor year expected after implementation of the SBO rule. As a result of the improvements made under the SBO rule, more plants achieved a lower SBO CDF than expected, and the plants with the greatest numbers of loss of offsite power from plant events and extremely severe weather conditions made the most improvement by providing access to an alternate ac power supply. In addition, maintaining high EDG target reliability levels provides assurance that probabilistic risk assessment/individual plant examination EDG performance assumptions are valid. With some exceptions, the observed EDG reliability performance generally exceeds the mean reliability EDG performance assumptions in the probabilistic risk assessment/individual plant examinations, indicating that SBO CDFs are smaller and better than stated in many probabilistic risk assessment/individual plant examinations. As the SBO rule risk reduction objectives have been exceeded, further investigation of strategies for reducing SBO frequencies (as suggested in NUREG–1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," December 1997) may not be needed.

- As a result of the SBO rule all plants have (1) established SBO coping and recovery procedures; (2) completed training for these procedures; (3) implemented modifications as necessary to cope with an SBO; and (4) ensured a 4- or 8-hour coping capability.

- Before the SBO rule was issued, only 11 of 78 plants surveyed had a formal EDG reliability program, 11 of 78 plants had a unit average EDG reliability less that 0.95, and 2 of 78 had a unit average EDG reliability of less that 0.90. Since the SBO rule was issued, all plants have established an EDG reliability program that has improved EDG reliability. A report shows that only 3 of 102 operating plants have a unit average EDG reliability less than 0.95 and above 0.90 considering actual performance on demand, and maintenance (and testing) out of service (MOOS) with the reactor at power. However, the analysis of EDG performance on demand indicates MOOS with the reactor at power is more than expected and can have a significant effect on the EDG reliability calculations. Increased MOOS explains why licensees

appear to be having difficulty meeting a 0.975 EDG target reliability. Decreased EDG reliabilities and/or increased MOOS unavailabilities erode the risk benefits obtained from implementing the SBO rule.

- The operating experience indicates that the SBO rule has increased defense-in-depth. The SBO related hardware and procedures have been used in response to unplanned events and provided additional protection. The SBO rule provides additional defense-in-depth to compensate for potential degradation of the ac offsite power system that may result from deregulation of the electric power industry or longer than expected recovery of offsite power following extremely severe weather conditions.

- A comparison of the value-impact expectations to the outcomes indicates that the value-impact was within the expected range of reductions in public dose-per-dollar of cost. As expected, there was wide variation in plant-specific values and impacts because the SBO rule allowed flexibility. Not expected was the addition of costly power supplies, which accounted for 75 percent of the estimated industry cost impact and explains why the NRC value-impact analysis underestimated the cost by a factor of 4. However, it appears licensees justified the cost of the power supplies by counting on offsetting monetary benefits, such as more operating flexibility from increased EDG allowed outage times. Thus the value was also underestimated. The remaining 25 percent of the estimated industry cost impact appears reasonable, considering the outcomes: known coping capabilities, industry risk reduction from plant-specific procedural and hardware enhancements, and additional defense-in-depth.

A comparison of the SBO rule expectations to the corresponding outcomes indicates that resolution of the generic issue of SBO was effective as no additional generic actions are warranted and no new generic safety issues have been identified.

Although the SBO rule was effective for the reasons stated above, consistent with adhering to Principles of Good Regulation that include clarity (coherent and practical regulations) and reliability (regulations based on operating experience) there are opportunities to revise the regulatory guidance and inspection documents. The proposed revisions are not intended to impose any new regulatory requirements, are consistent with the SBO technical basis (NUREG–1032); ensure high levels of EDG reliability; maintain present levels of safety by ensuring the risk benefits obtained from implementing the SBO rule do not erode; provide practical guidance for reactor shutdowns with limited offsite or onsite power sources; and use operating experience to improve the predictability and consistency of NRC decisions in the area of EDG reliability. The opportunities are as follows:

(1) Regulatory Guides 1.155, “Station Blackout,” August 1988, RG 1.9, “Selection, Design, Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants,” Rev. 3, July 1993, and RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2, March 1997, which address use of the existing EDG reliability terms, criteria, and measurements may need to be revised in a coherent manner to: (a) clarify that EDG unavailability due to MOOS with the reactor at power should be included in the reliability calculation, (b) clarify that licensees should balance increased EDG reliability against the increased EDG unavailability to maintain the RG 1.155 minimum individual EDG target reliabilities, (c) clarify that the EDG system boundary used in the reliability calculation should include the load sequencer and the bus between the EDG and the loads, and (d) establish common EDG start and load-run criteria for the guidance.

Inspection documents Temporary Instruction 2515/125, ?Inspection of Implementation of Station Blackout Rule," (no date), and Inspection Procedure 62706, "Maintenance Rule," December 31, 1997, may need revision to delete use of the Nuclear Management and Resources Council (NUMARC)(now NEI) trigger values to assess compliance with the 0.95 and 0.975 EDG target reliability. The NUMARC trigger values, which are not endorsed by the regulatory guidance, do not provide high levels of confidence that the EDG target reliability is being met; this is inconsistent with ensuring high EDG reliability, delays corrective action, and erodes the risk benefits obtained from implementing the SBO rule.

(2) Operating events indicate that the availability of some Aac power supplies is dependent on offsite or onsite power supplies. SBO-related inspection documents may need inspection attributes to verify that the Aac sources meet NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Revision 1, May 1993, Appendix B, B.8, "Minimal Potential For Common Cause Failure."

(3) Regulatory Guide 1.93, "Availability of Electric Power Sources," December 1974, the basis for technical specifications in the area of ac onsite and offsite power supply availability, provides for shutdown of the reactor after extended ac power supply unavailability. Plant shutdown with one or more offsite or onsite power supplies unavailable could exacerbate the grid condition or remove redundant sources to operate decay heat removal systems, increasing the likelihood of an SBO. Additional practical guidance may minimize the likelihood of an SBO.

(4) Follow-up at 2 plants found a large difference between the unit average EDG reliability based on load-run tests and unplanned demands and the reliability calculated by the licensee based on the last 100 start and load-run tests and unplanned demands. This difference confirms a previous finding in INEL-95/0035, "Emergency Diesel Generator Power System Reliability," February 1996, that the current testing and inspection activities (as prescribed by the NRC) may not be focusing on the dominant contributors to unreliability during actual demands. Accordingly, NRC inspection documents may need to be modified to better factor in the conditions and experiences gained from actual system demands to facilitate inspection of EDG compliance.

As lessons learned: (a) to the extent that the NRC staff revises existing regulatory documents to be more risk-informed and performance-based, they may need to be modified to ensure consistent interpretation and use of terms, goals, criteria, and measurements; and (b) new regulations or the accompanying regulatory documents should include quantitative objectives to facilitate evaluation of its regulatory effectiveness.

5 REFERENCES

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2. U.S. Nuclear Regulatory Commission, NUREG–1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," December 1997.

3. U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum, "Briefing on IPE Insight Report," May 28, 1997.

4. U.S. Nuclear Regulatory Commission, SECY–97-180, “Response to Staff Requirements Memorandum of May 28, 1997, Concerning Briefing on IPE Insight Report,” August 6, 1997.

5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.93, “Availability of Electric Power Sources,” December 1974.

6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.108, “Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants,” Revision 1, August 1977.

7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.9, “Selection, Design, Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants,” Revision 2, December 1979

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

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

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

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

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

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

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

15. U.S. Nuclear Regulatory Commission, RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” June 1993.

16. Nuclear Energy Institute, NUMARC 93-01, “Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plant,” May 1993.

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

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

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20. U.S. Nuclear Regulatory Commission, NUREG/CR-xxx, “Reliability Study Update: Emergency Diesel Generator Power System, 1987–1988 (DRAFT),” December, 1999.

21. U.S. Nuclear Regulatory Commission, NUREG/CR–5496, “Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1996,” November 1998.

22. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.93, “Availability of Electric Power Sources,” December 1974.

23. U.S. Nuclear Regulatory Commission, NUREG/CR–5994, “Emergency Diesel Generator: Maintenance and Failure Unavailability, and Their Risk Impacts,” October 1994.

24. U.S. Nuclear Regulatory Commission, RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2, March 1997.

25. Nuclear Energy Institute, NUMARC 93-01, “Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plant,” April 1996.

26. U.S. Nuclear Regulatory Commission, letter from James E. Taylor, Executive Director for Operations to J. Ernest Wilkins, Chairman, Advisory Committee on Reactor Safeguards, October 29, 1993.

27. U.S. Nuclear Regulatory Commission, RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” January 1995.

28. U.S. Nuclear Regulatory Commission, “Effects Of Electric Power Industry Deregulation on Electric Grid Reliability and Reactor Safety,” SECY–99-129, May 11, 1999.

29. U.S. Nuclear Regulatory Commission,”The Effects of Deregulation of the Electric Power Industry on the Nuclear Plant Offsite Power System: An Evaluation,” dated June 30, 1999.

30. U.S. Nuclear Regulatory Commission, “Offsite Power Voltage Inadequacies,” Information Notice 2000-06, March 27, 2000.

31. U.S. Nuclear Regulatory Commission, “Offsite Power Reliability Challenges From Industry Deregulation,” Information Notice 98-07, February 27, 1998.

32. U.S. Nuclear Regulatory Commission, Information Notice 97-21, “Availability of Alternate AC Power Source Designed for Station Blackout,” Page 21, April 18, 1997.

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

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. 33] 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. 34].

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. 35]. 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 because 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, that 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. 36] 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.

<u>Public and Industry Comment</u>

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. 37]. 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. 38], 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. 39], 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

(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. 40], 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. 41], 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. 42]), 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. 43]. 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. 44]. 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, that 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. 45]. 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. 46], 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 1560. In SECY–97-180 the staff also evaluated the industry's average cost per person-rem averted in satisfying the SBO requirements.

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38. Science and Engineering Associates, Inc., and Mathtech, Inc., "Response To Industry Comments on Station Blackout Cost Analysis (NUREG/CR–3840)," November 1986.

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40. U.S. Nuclear Regulatory Commission, "Final Station Blackout Rule, USI A-44," SECY–88-22, January 21, 1988.

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

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

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

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

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

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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						

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 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	
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				

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
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				
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

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
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				
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					

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	Shutdow n
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
Vogtle 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					

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
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					
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					

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
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					
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				

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
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					

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
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 SBO-CDF/Coping time in minutes/Aac access time in minutes
	PRA/IPE	Actual frequency/number of LOOP events at power since commercial operation	Approximate factor PRA/IPE underestimated in comparison to operating experience		
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 reliability 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)		
Requalify an EDG	$5.6M (2 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
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.

Operating Events

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.

Operating Events

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 incapable of being closed onto a de-energized bus. On 6/16/98, a similar test determined that the Gas Turbine No. 2 was also incapable 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 associated with 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.

Operating Events

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.

APPENDIX G

RESOLUTION OF EXTERNAL COMMENTS

Appendix G
Resolution of External Comments

An April 14, 2000 memorandum, “Draft Report, Regulatory Effectiveness of the Station Blackout Rule,” was sent to David Modeen, Director of Engineering, Nuclear Energy Institute from Charles E. Rossi, Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research [Ref. 1]. Identical letters were sent to other members of the industry and made publically available. The purpose of the April 14, 2000 memorandum was to obtain “peer review” comments before finalizing the report regarding (1) the reasonableness of the approach, (2) the appropriateness of the conclusions, (3) and other regulations, accompanying regulatory guides, and inspection documents that should be assessed to make U.S. Nuclear Regulatory Commission activities (NRC) more effective, efficient and realistic. Letters with comments were received from the Union of Concerned Scientists (UCS), the Nuclear Energy Institute (NEI), the Institute of Nuclear Power Operations (INPO), the Advisory Committee on Reactor Safeguards (ACRS), and the Electric Power Research Institute (EPRI).

Each organization providing comments is listed below, in order of the date received, followed by a restatement of their comments verbatim and the resolution of each comment. None of the peers reviewers provided comments to suggest other regulatory documents that should be assessed. Further, some comments were outside the scope of the report but were addressed for completeness. Italics were used to distinguish report revisions from the original text. Conforming changes were made throughout the report but are not listed below.

UNION OF CONCERNED SCIENTISTS (UCS) provided the following comments in a letter to the NRC dated April 20, 2000 [Ref. 2].

COMMENT 1: I fully agree with the general conclusion of this report that the station blackout rule and all of the attendant activities undertaken by the NRC and the industry have resulted in a significant, positive contribution to nuclear plant safety.

RESOLUTION: No response required

COMMENT 2: Having deep skepticism for probabilistic risk assessment as practiced by the NRC and the industry, I disagree with the stated numerical core damage frequency reductions resulting the SBO rule. For example, the ease with which PRA inputs and assumptions can be altered makes numerical comparisons like those reported in Table 3 on page 7 of the draft report absolutely meaningless. I agree that the SBO rule had the net effect of reducing core damage frequency.

RESOLUTION: Section 3.2.1 of the report was revised to clarify that the PRAs were altered by the licensees with a sound basis as follows: "Table 3, 'Probabilistic Risk Assessment/Individual Plant Examination Sensitivity Analyses,'" was prepared from information in plant-specific PRA/IPE sensitivity analyses to show the effect of typical SBO rule modifications on the overall risk. *Licensees modified PRA/IPEs based on plant specific hardware and procedural modifications related to SBO rule implementation. The modified PRA/IPEs consistently showed a reduction in the plant CDF, thus providing strong evidence for the risk reduction gained from promulgation of the SBO rule. ....*"

COMMENT 3: I cannot understand the information presented in Table 8. Some additional text explaining what the numbers in the rows and columns represents would be exceedingly helpful.

RESOLUTION: The comment is correct. Table 8 and its supporting paragraph were deleted from the report as they were deemed to be (a) too complex as a result of the use of uncommon terms and (b) unnecessary as the problem has been addressed. The information in Table 8 and the supporting paragraph called attention to severe weather related LOOPS having long recovery times. The same section of the report uses events to better illustrate that severe weather related LOOPs have had long recovery times; however, the industry has provided guidance for actions to be taken before the onset of severe weather.

COMMENT 4: The first bulleted paragraph on page x states that all plants have established SBO coping and recovery procedures that ensure a 4- or 8-hour coping duration. My experience in the industry before joining UCS strongly suggests that this conclusion is incorrect. In 1996, I worked as a consultant at the Haddam Neck nuclear plant on that licensee's response to an

NRC 50.54(f) letter relating to design and licensing bases configuration control. My specific assignment was to evaluate the design and licensing bases for station blackout. The licensee's decision in late 1996 not to restart the plant terminated my assignment before the final report was issued. However, in my work before that time, I determined that the plant's procedures and training complied with the NRC's regulatory guidance on SBO, but they did not comply with the rule itself. For example, all of the procedures used by the operations department at Haddam Neck in responding to a station blackout event assumed that the event began with the reactor at 100 percent power. If the plant was not at power, many of the procedural steps either could not be performed or would have made conditions worse. The exclusive focus of the procedures and training was on a SBO event from full power. If the SBO event occurred when the reactor was shut down for refueling with the entire reactor core offloaded into the spent fuel pool or shut down in mid-loop operation, the procedures and training were completely silent on what to do. Haddam Neck was not the only plant with this programmatic deficiency. The SBO rule does not limit the licensees' responsibilities to events at full power. The SBO rule requires licensees to be able to cope with an event lasting 4- or 8-hours. Until such time that the NRC staff verifies that all plants can indeed cope with a SBO event occurring at any time, not just when the reactor is at full power, the cited conclusion in the draft report is invalid.

I realize that Comment (4) may be outside the scope of your report. The report seems focused on what the SBO rule, not what it may not have done. In addition, the issues I raise in Comment (4) may be viewed as allegations. I am sending a copy of this letter to Mr. Ed Baker of the NRC staff for possible treatment as an allegation.

RESOLUTION: The UCS question will be addressed within the NRC's allegation process as Allegation No. NRR-2000-A-0020.

NUCLEAR ENERGY INSTITUTE (NEI) provided the following comments in a letter to the NRC dated June 14, 2000 [Ref. 3].

Comment 1: The approach used to assess the effectiveness of this rule appears to be reasonable. We suggest that the assessment focus on regulatory requirements as opposed to "expectations." The requirements of 10 CFR 50.63, *Loss of All Alternating Current Power*, should be the primary consideration.

RESOLUTION: Section 3.1 of the report was revised to address the comment as follows:

"For the purposes of this assessment, the regulatory documents present the expectations (desired outcomes) in terms of specific objectives, requirements, and guidance. *Hence, the requirements are an integral part of the set of 'expectations' which are collectively used as the basis for conducting an effectiveness review.*

The regulatory documents are considered effective if the expectations are being achieved. ..... *The expectations were established from objective measures stated in the SBO rule, RG 1.155, 53 FR 23203, and NUREG–1109 in the areas of coping capability, risk reduction, EDG reliability, and value-impact. The use of multiple regulatory documents provided an examination of the regulatory process."*

Comment 2: The conclusion that this rule provides "additional defense-in-depth to compensate for potential degradation of the ac offsite power system that may result from deregulation of the electric power industry" is not substantiated. Such statements are speculative at best and should not be included in the report. We believe the rule supports the conclusion in the report that the licensees have demonstrated a coping capability to deal with loss of offsite power consistent with the requirements of the station blackout rule.

RESOLUTION: Section 3.2.5 under Offsite Power System Deregulation was revised to be more robust as follows:

"*As stated in 53 FR 23203, it was expected that the SBO rule would provide additional defense in depth by requiring plants to cope with an SBO for a specified duration. The increment of of defense-in-depth is provided by the preventative effect of reducing the likelihood of core damage by implementing the SBO rule. In addition to increased coping capability, the SBO rule has resulted in plants gaining increased tolerance to an SBO from reduced risk as previously explained, which addresses the mitigative aspect of defense-in-depth. These SBO rule outcomes provide additional defense-in-depth to compensate for increased risks due to potential degradation of the ac offsite power system that may result from deregulation of the electric power industry, as explained in SECY–99-129, 'Effects Of Electric Power Industry Deregulation on Electric Grid Reliability and Reactor Safety,' May 11, 1999, a publicly available NRC technical report, 'The Effects of Deregulation of the Electric Power Industry on the Nuclear Plant Offsite Power System: An Evaluation,' dated June 30,*

1999; a recent Information Notice 2000-06, 'Offsite Power Voltage Inadequacies,' March 27, 2000, and Information Notice 98-07, 'Offsite Power Reliability Challenges From Industry Deregulation,' February 27, 1998."

Comment 3: The development of the trigger value concept for monitoring EDG target reliability involved extensive interactions between industry, NRC staff and ACRS. The NRC Office of Research contracted Dr. E. Lofgren, Science Applications International Corporation (SAIC), to conduct an independent review. His review confirmed the appropriateness of this trigger value concept as a mechanism for EDG target reliability. The trigger value concept was never intended to represent a robust statistical basis for individual EDG reliability.

The Commission issued a Staff Requirements Memorandum on SECY–90-340 dated June 28, 1991, that provided acceptance of the trigger value concept within the framework of the reliability levels assumed in the station blackout rule coping analysis.

RESOLUTION: That the trigger value concept was never intended to represent a robust statistical basis for individual EDG reliability is consistent with deleting these from the inspection procedures for the purposes of demonstrating achievement of the licensee's commitments to 0.95 or 0.975.

Appendix D, "Emergency Diesel Generator Trigger Values," of the report provides a history of the trigger values and includes the SAIC reports (NUREG–5078 and NUREG–5611), SRM, and SECY–90-340 stated in the comment. The SAIC reports provide statistical insights on multiple trigger values and trigger value combinations, and do not appear to be independent reviews.

Appendix D also indicates several other statistical studies, that applied multiple statistical methods, that were completed by several consultants. Some of these studies support use of the trigger value concept and some of these studies challenge the appropriateness of the trigger value concept. In addition, the main report and Appendix D explain that in a letter to the ACRS dated October 29, 1993, the Executive Director of Operations informed the ACRS that the staff agreed that conformance of individual EDGs with trigger values cannot be taken, in any statistical fashion, to mean that the EDG has demonstrated achievement of the licensee's commitments to 0.95 or 0.975. The letter stated that Note 3 was added to RG 1.160, June 1993 to emphasize this fact. The main report also states that industry documents that use the EDG trigger values have not been endorsed and that wording in earlier and now outdated NRC regulatory guides may have caused confusion about use of the trigger values.

Also see ACRS comment 3 and ERPI comments 2 and 4.

COMMENT 4: We support the concept of ensuring the regulatory documents are coherent and consistent with regulation. New issues that may arise during implementation should be

addressed within the appropriate guidance documents. However, proposed modifications to regulatory documents should not impose new requirements beyond what is stated in the regulations. If NRC concludes that new requirements or clarifications to existing requirements are necessary, then rulemaking would be warranted.

RESOLUTION: Rulemaking would used be for new requirements or clarifications that add new requirements, however the proposed clarifications are not intended to impose any new regulatory requirements. The report conclusions were revised to state that, "*Although the SBO rule was effective for the reasons stated above,* consistent with adhering to Principles of Good Regulation that include clarity (coherent and practical regulations) and reliability (regulations based on operating experience) there are opportunities to revise the regulatory *guidance and inspection* documents. *These revisions are not intended to impose any new regulatory requirements*, are consistent with the SBO technical basis (NUREG–1032); ensure high levels of EDG reliability; *maintain present levels of safety by ensuring the risk benefits obtained from implementing the SBO rule do not erode*; provide practical guidance for reactor shutdowns with limited offsite or onsite power sources; and use operating experience *to improve the predictability and consistency of NRC decisions in the area of EDG reliability*. The opportunities are as follows as stated in the report recommendations:

'(1) Regulatory Guides 1.155.... which address use of *existing* EDG reliability terms, criteria, and measurements may need to be revised in a coherent manner to:...';

'Inspection documents....The NUMARC trigger values, *which are not endorsed by the regulatory guidance*, do not provide high levels of confidence that the EDG target reliability is being met, ...'

'(4) Follow-up at 2 plants....Accordingly, NRC inspection documents may need to be modified to better factor in the conditions and experiences gained from actual system demands to *facilitate inspection of EDG reliability compliance.'"*

In addition, the report was planned to be consistent with the strategies for implementing NRC strategic performance goals in the areas of maintaining safety; reducing the unnecessary regulatory burden; making NRC activities effective, efficient, and realistic; and public confidence. The report was revised where appropriate demonstrate its consistency with the strategic performance goals.

ADVISORY COMMITTEE ON REACTOR SAFETY GUARDS (ACRS) provided comments in a letter dated June 22, 2000[Ref. 4], following a June 7, 2000, presentation by RES to the ACRS in a public forum. At this meeting, RES committed (1) to revise Table 6 outcomes for the public does reduction and the value impact ratio and (2) revise the conclusions of the report to add a lesson learned that *"In addition, new regulations or the accompanying regulatory documents should include quantitative regulatory objectives to facilitate evaluation of its regulatory effectiveness*."

The ACRS letter comments and their resolution are as follows:

COMMENT 1: The initiative undertaken by the staff to evaluate selected regulations to determine whether they have been effective in achieving their objectives is valuable and should be continued.

RESOLUTION: RES plans to issue its review of the anticipated transient without scram (ATWS) rule for internal and external peer review in FY 2000; and in FY 2001 finalize reports on ATWS, Appendix J, and one other topic to be named upon completion of discussions with other offices. Two reviews are presently planned through FY 2002.

COMMENT 2: Regulatory documents related to the Station Blackout (SBO) rule should be reviewed to eliminate identified inconsistencies in the definition of reliability. Because of these inconsistencies, the intended reliability targets for emergency diesel generators (EDGs) are not being met in some cases.

RESOLUTION: We agree with this comment. RES plans to: (a) clarify that EDG unavailability during maintenance or test with the reactor at power should be included in the reliability calculation,(b) clarify that licensees should balance increased EDG reliability against the increased EDG unavailability to maintain the RG 1.155 minimum individual EDG target reliabilities, (c) clarify that the EDG system boundary used in the reliability calculation should include the load sequencer and the bus between the EDG and the loads, and (d) establish common EDG start and load-run criteria for the guidance.

It should be noted that the SBO report focused on the SBO rule and related SBO regulatory documents and did not address plant-specific issues. A sampling of plants that have not met EDG reliability targets have been identified in NUREG/CR–5500, Volume 5, INEL-95/0035, "Emergency Diesel Generator Power System Reliability," February 1996. This report is being updated by RES to include more plants and current operating experience. Upon completion of the above activities, RES will determine the course of action to be taken to address the findings of the updated NUREG/CR–5500.

COMMENT 3: Acceptance of the use of trigger values in inspection documents should be discontinued.

RESOLUTION: Comment 3 is consistent with the conclusions of the report. The Office of Nuclear Reactor Regulation will revise the relevant inspection procedure guidance to discontinue acceptance of licensee use of EDG trigger values.

COMMENT 4: The evaluation of the regulatory effectiveness of the SBO rule provides significant lessons that should be beneficial in preparing a template for the evaluation of other regulations and in the development of future regulations.

RESOLUTION: The ACRS suggested preparation of a template for future regulatory effectiveness evaluations using the lessons learned from the SBO rule evaluation. This suggestion has been implemented as the methods and lessons learned from the SBO evaluation are being used as a template for other regulatory effectiveness evaluations. The ACRS also observed that the report shows the importance of establishing a risk-reduction expectation before development of a new regulation. We agree that development of the risk-reduction expectation early in the development of the rule is a good approach and is consistent with our activities in risk-informing Part 50. As you are aware, the current regulatory process still ensures that a substantial improvement in safety is demonstrated as outlined in NUREG/BR–0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,” Revision 2, dated November 1995.

INSTITUTE OF NUCLEAR POWER OPERATIONS (INPO) provided comments in a letter to the NRC dated July 14, 2000 [Ref. 5].

GENERAL COMMENT: The approach and conclusion of the report appear reasonable.

RESOLUTION: No response required

COMMENT 1: We recommend that the specific core damage frequency (CDF) numbers contained in the report be removed since they are characterized as approximations. It would be sufficient to say reductions in CDF are supported by CDF calculations.

RESOLUTION: Section 3.1 of the report was revised as follows: "When using the above PRA/IPEs it should be recognized that their data *are estimates* and do not always reflect the current design or operating performance of safety systems."

COMMENT 2: The conclusion that the value impact was within the expected range of reductions in public dose-per-dollar of cost is not well supported. The justifying statement, "However, it appears licensees justified the cost of the power supplies by counting on offsetting monetary benefits from increased EDG allowed outage times," is not supported by any monetary cost/value data.

RESOLUTION: Section 3.2.4 of the report states the cost as follows: "The cost of an additional EDG at Davis Besse was $9.07M. This value was taken as a representative amount for the addition of power supplies and used in the cost estimates shown below." Section 3.2.4 also states the value as follows: "In submitting this change to the NRC, Davis Besse noted that its SBO diesel generator installation would save $5.25M over the remaining life of the plant, including $3.15M for increased flexibility in performing maintenance on its safety EDGs and $2.1M in replacement power costs."

COMMENT 3: The opportunities identified to improve regulatory documents do not support the main conclusion of the report. For example, the report concludes that emergency diesel generator (EDG) reliability has improved. However, the report also suggests that NRC test and inspection documents have not focused sufficiently on dominant contributors to EDG unreliability during actual demands. Justification should be provided on why the identified opportunities do not affect the main conclusions of the report.

RESOLUTION: See response to NEI comment 4.

ELECTRIC POWER RESEARCH INSTITUTE (EPRI) provided comments in a letter to the NRC dated July 24, 2000[Ref. 6].

Comment 1: Regarding the NRC draft report, we agree that the SBO rule, along with other industry programs (e.g. INPO's trip reduction program, installation of risk meters) has contributed to improving safety at nuclear units, including greater tolerance of loss of offsite/onsite ac power. We also agree that the SBO reduction objectives have been exceeded, and as a consequence that major new programs are not needed. We do not feel, however, that some of the revisions to the associated regulatory guides, as proposed in the report, would necessarily improve the safety of nuclear plants. At the current time, revisions probably could not be shown to be cost/beneficial, and the current direction of the NRC and industry toward Risk Informed Regulation, it may be better to maintain the current technical approach as is for the time being. The implementation of Configuration Risk Management Programs at many plants, which can help deal with degraded ac offsite/onsite power situations, further underscores this conclusion.

RESOLUTION: See the response to NEI comment 4 .

Comment 2: One particular reservation we have in reviewing the draft report concerns changing the failure rate triggers used in the SBO Rule. At the time of their adoption, extensive technical analyses, including Monte Carlo studies, were performed to justify their basis. These analyses recognized that, when dealing with small statistical samples involving highly reliable machines, high levels of confidence are impossible to achieve, without creating large numbers of "false negatives." These would inadvertently involve utilities in costly an unnecessary remediation programs. When the current failure rate triggers were established, the technical people (including EPRI and the NRC staff, and contractors involved took into account the difficult problem of balancing the need for high confidence with the very large number of events needed to obtain this confidence. The report proposes to delete the NUMARC triggers but does not present an alternative. We encourage NRC to review this section of the report and re-examine its technical basis in light of the amount of information necessary to make important regulatory decisions when data will be sparse. Further details are provided in Attachment A.

RESOLUTION: See NEI comment 3 and ACRS Comment 3. Attachment A was treated as a EPRI comment 4, below.

Review of the operating experience, the inspection reports, and INEL-95/0035 found only two indications of repetitive EDG failures within a site over a 12 year period of time. The real situation appears to be that licensee complete root cause investigations and take actions to prevent recurrence of problems (per 10 CFR Part 50, Appendix B, XVI, "Corrective Action"); deferring these activities based on statistically-variation-induced false alarms, consideration of a false alarm rate (percentage of the time the true EDG reliability is observed) does not appear to be a realistic concern due to the strength of the root cause/corrective action process. In addition, this is not consistent with maintaining high levels of EDG reliability, or

common industry practice to take timely corrective action as well as action to prevent recurrence of a problem .

Comment 3: In conclusion, let me once again emphasize our point of agreement: the SBO rule has been effective in helping improve safety, and that major changes are not appropriate at this point in time.

RESOLUTION: See response to NEI comment 4.

Comment 4: This was presented as Attachment A, “EPRI Comments on Target Reliability and Trigger Values,” as restated verbatim below.

Between 1988 and 1991, EPRI carried out the analytical work underlying the failure trigger rate methodology. NUMARC, NUGSBO and NRC subsequently adopted this methodology. Both industry and NRC would like a way of knowing that an EDG whose target reliability is 95% has a 95% confidence of being above the target value. Manifestly, if one had a sample of 1000 demands to evaluate, such assurance could be approached.

However the real situation is that EDG demands are accrued relatively slowly – as little as one per month. It may take almost two years to experience 20 demands, 4 years for 50 demands, and 8 years for 100 demands. Unfortunately, one cannot wait for 100 demands to determine whether an EDG’s reliability is acceptable. Thus it was decided to also assess EDG acceptability based on much smaller samples, namely 50 and 20 demands.

When the sample size is small, and when one is dealing with target reliabilities at the very high end of the reliability range, it is statistically impossible (even with no observed failures) to provide high confidence that the actual reliability equals or exceeds the target reliability. For example, using binomial distribution for 20-demand sample illustrates the dilemma. Even if zero failures were permitted, one would know with 95% confidence that the underlying reliability is only greater than 83%. If one failure were permitted, one would only know with 95% confidence that the underlying reliability is greater that 75%. One would need close to 75 demands with zero failures in order to be able to be 95% confident that 95% reliability had been achieved.

Unfortunately, the above-described statistical limitations are not the most troublesome problem that derives from the need to use small samples to estimate reliability. The creation of statistical -variation-induced false alarms that accompanies high confidence that target reliabilities are met is real world and expensive. Each false alarm requires that an EDG enter a corrective action remedial program. For example, if in order to achieve high confidence that target reliability is met, zero (or even one) failure in 20 demands is allowed, an EDG may end up in a perpetual reliability program–due to perpetual false alarms. Large sample sizes would be of reduced importance because the EDGs would be “false-alarmed-out.” There would be no value to these EDG remedial programs.

RESOLUTION: Appendix D, “Emergency Diesel Generator Trigger Values,” of the report was revised to use the information provided Attachment A of this comment as follows:

. *....In practice, many licensees monitor EDG reliability in the short term (20 demands) and long term (50, 100 demands) as well as a unit average EDG reliability based on the same number of demands.......*

.....The individual EDG target reliabilities of 0.95 or 0.975 are statistically reasonable and practical so long as there is sufficient data to determine EDG reliability with confidence (typically 50 to 95 percent). However individual EDG demands accrue relatively slowly, typically about 20–25 per fuel cycle, limiting the amount of data that can be collected in one fuel cycle. Obviously but one failure of an EDG with a 0.975 target reliability with less than 40 demands results in an unfavorable statistic for a typical fuel cycle. When failures occur with small amounts of data, statistical methods are generally used to help interpret the data. The NRC has completed the following statistical that resulted in a trigger value concept to help interpret EDG failure data and make decisions as summarized below......

......The NRC, the industry, and their consultants have applied the above statistical methods to facilitate the interpretation of the EDG reliability data. Some of these methods resulted in trigger values based on statistically-variation-induced false alarms to account for the chance that the failure may be the result of a non-valid demand or to account for other possibilities for not observing the true reliability. The concern is that one would not want to enter a costly corrective action program based on a false statistic.

In practice, the acceptability of an EDG’s reliability is based on analysis. Following a failure licensees complete root cause investigations and take corrective actions and actions to prevent recurrence of a problem consistent with 10 CFR 50, Appendix B, XVI, “Corrective Action.” Consequently licensees would enter a corrective action upon determination of a real problem. In addition, should this investigation find that there was no valid demand or failure, it would not be reflected in the statistics. It appears statistically-variation-induced false alarms or concerns about entering a costly corrective program unnecessarily are not practical considerations.

REFERENCES:

1. Memorandum from Charles E. Rossi, Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, to David Modeen, Director of Engineering, Nuclear Energy Institute,"Draft Report, Regulatory Effectiveness of the Station Blackout Rule," April 14, 2000.

2. Memorandum from David A. Lochbaum, Union of Concerned Scientists to Charles E. Rossi, Director, Division of Systems Analysis and Regulatory Effectiveness, U.S. Nuclear Regulatory Commission, "Comments on Draft Report on Station Blackout Rule," April 20, 2000.

3. Memorandum from Alex Marion, Nuclear Energy Institute to Charles E. Rossi, Director, Division of Systems Analysis and Regulatory Effectiveness, U.S. Nuclear Regulatory Commission, "Draft Report, 'Regulatory Effectiveness of the Station Blackout Rule,'" June 14, 2000.

4. U.S. Nuclear Regulatory Commission memorandum from Dana Powers, Chiarman, Advisory Committee on Reactor Safeguards, to Dr. William D. Travers, Executive Director for Operatons, "Draft Report, 'Regulatory Effectiveness of the Station Blackout Rule,'"June 22, 2000.

5. Memorandum from Mark A Peifer, Institute of Nuclear Power Operations to Dr. Farouk Eltawila, Acting Division Director, Division of Systems Analysis and Regulatory Effectiveness, U.S. Nuclear Regulatory Commission, "Draft Report, 'Regulatory Effectiveness of the Station Blackout Rule,'" July 14, 2000.

6. Memorandum fromJ.J. Haugh, Electric Power Research Institute, to Charles E. Rossi, Director, Division of Systems Analysis and Regulatory Effectiveness, U.S. Nuclear Regulatory Commission, "NRC's DRAFT Station Blackout Report," July 24, 2000.