Attachment 1

SAIC-91/6656

TECHNICAL EVALUATION REPORT
MONTICELLO NUCLEAR GENERATING PLANT
STATION BLACKOUT EVALUATION

TAC No. 6B569

Science Applications International Corporation
An Employee-Owned Company

Final

April 19, 1991

Prepared for:

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Contract NRC-03-87-029
Task Order No. 38

9104260170

1710 Goodridge Drive, P.O. Box 1303, McLean, Virginia 22102 (703) 821-4300
Other SAIC Offices: Albuquerque, Boston, Colorado Springs, Dayton, Huntsville, Las Vegas, Los Angeles, Oak Ridge, Orlando, Palo Alto, San Diego, Seattle, and Tucson

TABLE OF CONTENTS

TECHNICAL EVALUATION REPORT
MONTICELLO NUCLEAR GENERATING PLANT
STATION BLACKOUT EVALUATION

1.0 BACKGROUND

On July 21, 1988, the Nuclear Regulatory Commission (NRC) amended its regulations in 10 CFR Part 50 by adding a new section, 50.63, "Loss of All Alternating Current Power" (1). The objective of this requirement is to assure that all nuclear power plants are capable of withstanding a station blackout (SBO) and maintaining adequate reactor core cooling and appropriate containment integrity for a required duration. This requirement is based on information developed under the commission study of Unresolved Safety Issue A-44, "Station Blackout" (2-6).

The staff issued Regulatory Guide (RG) 1.155, "Station Blackout," to provide guidance for meeting the requirements of 10 CFR 50.63 (7). Concurrent with the development of this regulatory guide, the Nuclear Utility Management and Resource Council (NUMARC) developed a document entitled, "Guidelines and Technical Basis for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC 87-00 (8). This document provides detailed guidelines and procedures on how to assess each plant's capabilities to comply with the SBO rule. The NRC staff reviewed the guidelines and analysis methodology in NUMARC 87-00 and concluded that the NUMARC document provides acceptable guidance for addressing the 10 CFR 50.63 requirements. The application of this method results in selecting a minimum acceptable SBO duration capability from two to sixteen hours depending on the plant's characteristics and vulnerability to station blackout. The plant characteristics affecting the required coping capability are: the redundancy of the onsite emergency AC power sources, the reliability of onsite emergency power sources, the frequency of loss of offsite power (LOOP), and the probable time to restore offsite power.

In order to achieve consistent systematic responses from licensees to the SBO rule and to expedite the staff review process, NUMARC developed two generic response documents. These documents were reviewed and endorsed (9) by the NRC staff for the purposes of plant-specific submittals. The documents are titled:

1. "Generic Response to Station Blackout Rule for Plants Using Alternate AC Power," and

2. "Generic Response to Station Blackout Rule for Plants Using AC Independent Station Blackout Response Power."

A plant-specific submittal, using one of the above generic formats, provides only a summary of results of the analysis of the plant's station blackout coping capability. Licensees are expected to ensure that the baseline assumptions used in NUMARC 87-00 are applicable to their plants and to verify the accuracy of the stated results. Compliance with the SBO rule requirements is verified by review and evaluation of the licensee's submittal and audit review of the supporting documents as necessary. Follow up NRC inspections assure that the licensee has implemented the necessary changes as required to meet the SBO rule.

In 1989, a joint NRC/SAIC team headed by an NRC staff member performed audit reviews of the methodology and documentation that support the licensees' submittals for several plants. These audits revealed several deficiencies which were not apparent from the review of the licensees' submittals using the agreed upon generic response format. These deficiencies raised a generic question regarding the degree of the licensees' conformance to the requirements of the SBO rule. To resolve this question, on January 4, 1990, NUMARC issued additional guidance as NUMARC 87-00 Supplemental Questions/Answers (10) addressing the NRC's concerns regarding the deficiencies. NUMARC requested that the licensees send their supplemental responses to the NRC addressing these concerns by March 30, 1990.

2.0 REVIEW PROCESS

This review of the licensee's submittal is focused on the following areas consistent with the positions of RG 1.155:

A. Minimum acceptable SBO duration (Section 3.1),

B. SBO coping capability (Section 3.2),

C. Procedures and training for SBO (Section 3.4),

D. Proposed modifications (Section 3.3), and

E. Quality assurance and technical specifications for SBO equipment (Section 3.5).

For the determination of the proposed minimum acceptable SBO duration, the following factors in the licensee's submittal are reviewed: a) offsite power design characteristics, b) emergency AC power system configuration, c) determination of the emergency diesel generator (EDG) reliability consistent with NSAC-108 criteria (11), and d) determination of the accepted EDG target reliability. Once these factors are known, Table 3-8 of NUMARC 87-00 or Table 2 of RG 1.155 provides a matrix for determining the required coping duration.

For the SBO coping capability, the licensee's submittal is reviewed to assess the availability, adequacy and capability of the plant systems and components needed to achieve and maintain a safe shutdown condition and recover from an SBO of acceptable duration which is determined above. The review process follows the guidelines given in RG 1.155, Section 3.2, to assure:

a. availability of sufficient condensate inventory for decay heat removal,

b. adequacy of the Class 1E battery capacity to support safe shutdown,

c. availability of adequate compressed air for air-operated valves necessary for safe shutdown,

d. adequacy of the ventilation systems in the vital and/or dominant areas that include equipment necessary for safe shutdown of the plant,

e. ability to provide appropriate containment integrity, and

f. ability of the plant to maintain adequate reactor coolant system inventory to ensure core cooling for the required coping duration.

The licensee's submittal is reviewed to verify that required procedures (i.e., revised existing and new) for coping with SBO are identified and that appropriate operator training will be provided.

The licensee's submittal is reviewed for any proposed modifications to emergency AC sources, battery capacity, condensate capacity, compressed air capacity, appropriate containment integrity and primary coolant make-up capability. Technical specifications and quality assurance requirements set forth by the licensee to ensure high reliability of the equipment, specifically added or assigned to meet the requirements of the SBO rule, are assessed for their adequacy.

This SBO evaluation is based on a review of the licensee's submittals dated April 17, 1989 (12), June 16, 1989 (20), October 17, 1989 (13) and March 29, 1990 (14), the licensee's responses (15 and 16) to the question raised during the review and a telephone conversation on January 15, 1991 between the NRC/SAIC and the licensee staff, and the available information in the plant Updated Safety Analysis Report (USAR) (17); it does not include a concurrent site audit review of the supporting documentation. Such an audit may be warranted as an additional confirmatory action. This determination would be made and the audit would be scheduled and performed by the NRC staff at some later date.

3.0 EVALUATION

3.1 Proposed Station Blackout Duration

Licensee's Submittal

The licensee, Northern States Power (NSP), calculated (12 and 13) a minimum acceptable station blackout duration of four hours for the Monticello Nuclear Generating Plant. The licensee stated that no modifications are necessary to attain this proposed coping duration.

The plant factors used to calculate the proposed SBO duration are:

1. **Offsite Power Design Characteristics**

   The plant AC power design characteristics group is "P1" based on:

   a. Expected frequency of grid-related LOOPs of less than one per 20 years,

   b. Estimated frequency of LOOPs due to extremely severe weather (ESW) which places the plant in ESW Group "1,"

   c. Estimated frequency of LOOPs due to severe weather (SW) which places the plant in SW Group "2," and

   d. Independence of the plant offsite power system characteristic of "I1/2."

2. **Emergency AC (EAC) Power Configuration Group**

   The EAC power configuration group at Monticello is "C." The site is equipped with two emergency ac power supplies, one of which is necessary to operate safe shutdown equipment following a LOOP.

3. Target Emergency Diesel Generator Reliability

The licensee stated that a target EDG reliability of 0.950 was selected based on the unit average EDG reliability for the last 100 demands of greater than 0.950, consistent with NUMARC 87-00. In a later submittal (14) the licensee indicated its intent to maintain that target.

Review of Licensee's Submittal

Factors which affect the estimation of the SBO coping duration are: the independence of the offsite power system grouping, the estimated frequency of LOOPs due to ESW and SW conditions, the expected frequency of grid-related LOOPs, the classification of EAC, and the selection of EDG target reliability.

Our review of the plant USAR indicates that safeguards equipment has three offsite supplies of electrical power, the No. 1 reserve transformer (1R), the primary station auxiliary transformer (2R) and the auxiliary reserve transformer (1AR) (Figure 1). Each power source can be connected to both safeguards buses. In addition:

1. All offsite power sources are connected to the plant through electrically connected switchyards,

2. The safeguard buses are normally powered from offsite power sources through primary Station Auxiliary Transformer 2R, and

3. Upon a loss of power for this transformer, the safeguard buses are powered by No. 1 reserve transformer through an automatic transfer.

Based on this, the Monticello offsite power characteristics is classified as "I2," per guidance provided in RG 1.155, Table 5.

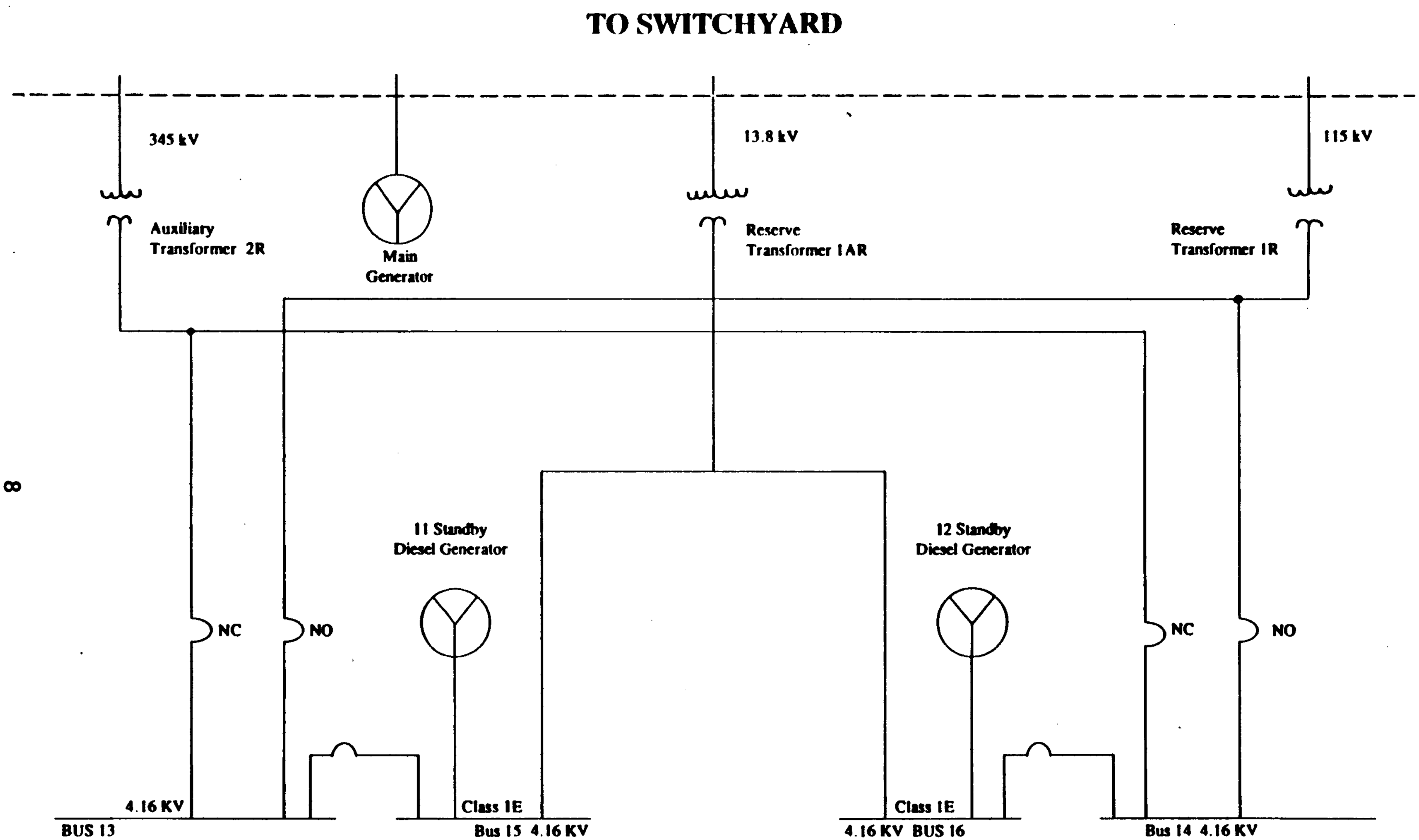

Figure 1. Monticello Offsite /Onsite Electrical Connections

The licensee's estimation of the extremely severe weather group for Monticello as "1" is in agreement with Table 3-2 of NUMARC 87-00, and is appropriate. Using Table 3-3 of NUMARC 87-00, the expected frequency of LOOPs at Monticello due to SW condition is estimated to be "0.0157" or "0.0085" depending on the site having offsite power transmission lines either on one or multiple rights-of-way, respectively. These values place Monticello in SW group "3" and "2" respectively. Review of the Monticello USAR indicates that the site has transmission lines on multiple rights-of-way, hence SW group "2," as claimed by the licensee, is appropriate.

With regard to the expected frequency of grid-related LOOPs at the site, we can not confirm the stated results. The available information in NUREG/CR-3992 (3), which gives a compendium of information on the loss of offsite power at nuclear power plants in U.S., indicates that Monticello did not have grid-related LOOP up to 1984. In the absence of any contradicting information, we agree with the licensee's statement that the frequency of grid-related LOOPs is expected to be less than one per 20 years.

Monticello has two emergency AC power sources powering two Class 1E safety buses, of which one is needed to supply safe shutdown loads following a LOOP, hence the licensee correctly identifies this configuration as "C."

The licensee stated that a target EDG reliability of 0.950 has been selected based on the demonstrated unit average EDG reliability for the last 100 demands, and that this target reliability would be maintained in accordance with Appendix D of NUMARC 87-00. Although this is an acceptable criterion for choosing an EDG target reliability, the guidance in RG 1.155 requires that the EDG reliability statistics for the last 20 and 50 demands also be calculated. Without this information it is difficult to judge how well the EDGs have performed in the past and if there should be a concern. The available information in the NSAC-108, which gives the EDG reliability data at U.S. nuclear reactors for calendar years 1983 to 1985, indicates that the EDGs at Monticello experience an average of 16 start demands per EDG per year with no failure reported. Using this data, it appears that the EDG target reliability (0.95) selected by the licensee (12) is appropriate. Nevertheless, the licensee needs to have an analysis showing the EDG reliability statistics for the last 20, 50, and 100 demands in its SBO submittal supporting documents.

Utilizing the above factors in Table 3-5a of NUMARC 87-00 results in an offsite power design characteristics of "P1," which leads to a required coping duration of four hours from Table 3-8, confirming the licensee's submittal.

3.2 Station Blackout Coping Capability

The plant coping capability with an SBO event for the required duration of four hours is assessed based on the following results:

1. Condensate Inventory for Decay Heat Removal

   Licensee's Submittal

   The licensee stated that 76,540 gallons of water are required for decay heat removal during the four hours of an SBO event. The calculation was based on Section 7.2.1 of NUMARC 87-00. The licensee provided (15) a breakdown of the required condensate inventory as: 36,940 gallons for decay heat removal, 33,600 gallons for recirculation pump seal leakage (70 gpm per pump, according to the USAR), and 6,000 gallons for the maximum allowed technical specification leakage. The minimum permissible condensate storage tank level, per Technical Specifications, provides 75,000 gallons of water. The suppression pool with a minimum capacity per Technical Specifications of 508,674 gallons is available as the alternate water source. The plant does not require (15) depressurization or cooldown of the reactor during an SBO.

Review of Licensee's Submittal

Using NUMARC methodology and a maximum reactor power of 1670 MWt, the plant would require 36,940 gallons of condensate to remove decay heat during a four hour SBO event. In addition, condensate is required to replenish reactor coolant system (RCS) inventory losses due to an expected recirculation pump seal leakage, and the technical specification maximum allowed leakage during an SBO. The licensee's assumption of a 70 gpm recirculation pump seal leak rate is conservative when compared to the NUMARC guidance, but is consistent with the licensee's USAR document. The licensee stated that this leak rate assumption is consistent with the results of a 1978 General Electric (GE) topical report, NEDO-24083 (18). We performed a cursory review of the GE document and concur with the licensee's assumption.

With regard to the suppression pool temperature, we performed an adiabatic heat-up calculation assuming that only decay heat will be deposited to the suppression pool during the four-hour SBO event. The heat from the RCS leakage was assumed to be released in the drywell. Using an initial suppression pool temperature of 90°F, we estimated that the final temperature would be approximately 166°F. Since this analysis does not consider any heat transfer from the suppression pool water to the torus wall, the licensee's statement that cooldown or depressurization is not needed appears to be reasonable. The suppression pool temperature can be as high as 145°F before the reactor pressure needs to be decreased below 1000 psig. Therefore, we conclude that depressurization or cooldown is not required, and the plant has adequate condensate inventory to cope with an SBO of four-hour duration.

2. Class 1E Battery Capacity

Licensee's Submittal

A battery capacity calculation has been performed that verifies that the Class 1E batteries have sufficient capacity to meet SBO loads for four hours at Monticello. The licensee stated (15) that the only load shed would be the emergency lighting, which would be disconnected after 30 minutes. Access and egress emergency lighting, and the control room emergency lighting are provided by self-contained battery pack powered lights which are sized for eight hours as part of Appendix R requirements. During the telephone conversation on January 15, 1991, the licensee stated that the adequacy of lighting to meet 10 CFR 50 Appendix R requirements was examined as part of the audit of emergency operating procedures.

Review of Licensee's Submittal

The 125 and 250 volt battery loads presented in the USAR and the capacity of the batteries were examined to determine if sufficient capacity exists to support the loads of a four hour SBO event. Prior to accepting the licensee proposal to shed emergency lighting loads, the licensee needs to confirm that the Appendix R lighting is adequate to support the control room operations under SBO conditions. The plant USAR indicates, although it does not specifically state, that the batteries have sufficient capacity to last for four hours withc· being charged.

An NRC Inspection Report on the electrical distribution system at Monticello in December, 1990 (Reference 19), found that the 250 volt battery minimum voltage had been miscalculated. Adjusting the minimum voltage upward, as required, would have the effect of reducing the battery capacity. The inspection report identified that the minimum voltage of the Division I and Division II 250 volt battery should be 216 and 218 volts, respectively. These values are required to ensure that the Class 1E instrumentation power supply inverters have at least a minimum voltage of 210 VDC. The licensee needs to verify that a minimum cell voltage of 1.80 and 1.82 VDC was considered in the Division I and II 250 volt battery sizing calculations, respectively.

The licensee stated (13) that NUMARC 87-00 Section 7.2.2 was used to verify the battery capacity. By this statement we consider that the licensee used IEEE Std-485 and therefore included an ageing correction factor of 1.25 as recommended by the standard. The licensee needs to verify that this ageing correction factor has been used, or needs to provide justification for using a smaller correction factor.

3. Compressed Air

Licensee's Submittal

The licensee stated (12) that transfer of the power supply for the alternate safety relief valve nitrogen supply to the Class 1E inverters is necessary to ensure that air operated valves required for decay heat removal during an SBO of four hour duration have sufficient backup sources for operation.

The licensee's October 1989 submittal (Reference 13) added a second modification to ensure operation of air operated valves:

> Install a nitrogen supply to the air valves which drain condensate from the steam supply line to the High Pressure Coolant Injection (HPCI) turbine and the steam exhaust line from the HPCI turbine.

On March 29, 1990, the licensee, in a supplemental response (14), stated that the installation of a nitrogen supply to the air operated valves that drain condensate from the steam supply and steam exhaust lines of the HPCI turbine was no longer necessary due to additional analysis.

Review of Licensee's Submittal

The HPCI system and the reactor pressure vessel safety relief valves were reviewed to determine their dependency on compressed air. No requirements, other than the transfer of the power supply for the nitrogen of the alternate safety relief valve, which was identified by the licensee, were found.

The licensee was questioned regarding the reversal in position about need to install nitrogen supply for the drains of the HPCI steam supply and exhaust. In Reference 15, Item 4, the licensee stated that alternate paths for condensate had been found. This explanation is found adequate.

4. Effects of Loss of Ventilation

Licensee's Submittal

The dominant areas of concern (DACs) at Monticello are listed in the following table along with their associated station blackout temperature, type of heat-up analysis performed, and justification for Reasonable Assurance of Operability (RAO).

AREA	STEADY STATE TEMP.	ANALYSIS	RAO JUSTIFICATION
HPCI pump pump room	151°F	NUMARC	equipment evaluation
Control Room	109.7°F	Non-NUMARC, see ref 16	less than 120°F
Drywell	270°F	Non-NUMARC, see ref 16	equipment evaluations
Torus Room	146°F	Qualitative estimate as discussed in Reference 16	equipment evaluations

Reasonable assurance of equipment operability is established without further analysis if temperatures in the DAC are calculated to be equal to or less than 120°F (NUMARC 87-00 Supplemental Questions and Answers #2.2) (10).

The licensee claims that the two SBO response valves in the steam tunnel, the HPCI Steam Supply Outboard Containment Isolation Valve (MO-2035) and the HPCI Discharge to Feedwater Piping (MO-2068), are not required to operate at the elevated temperature resulting from a loss of ventilation. These two valves open upon the first initiation of HPCI, which occurs during the initial portion of an SBO, and remain open. No further operation of these valves is required during an SBO, therefore the steam tunnel is not an area of concern and no temperature calculation was provided.

Review of Licensee's Submittal

The licensee provided (16) a brief description of the control room heat-up calculation. Upon the review of this summary we have identified the following concerns:

a. The licensee used a heat generation rate of 250 BTU/hr per occupant. This is a factor of three less than that recommended by the ASHRAE handbook (21), which is 230 watts per person.

b. The licensee assumed an initial control room temperature of 75°F. This value would be reasonable for a control room with two redundant HVAC trains, if there was a technical specification which required that at least one train always be operable. Otherwise, the licensee would need to consider a higher initial control room temperature consistent with historical maximum control room temperature without HVAC.

The drywell temperature rise was calculated by the licensee using the "Modular Accident Analysis Program" (MAAP), resulting in an estimated drywell temperature of 270°F. The licensee assumed a primary system leakage corresponding to a small break LDCA with a leak rate of 165 gpm. The licensee did not provide details of this analysis for review, however, based on this large primary system leakage, the drywell temperature estimate seems to envelope the expected SBO conditions.

The licensee's assumption that the steam tunnel temperature rise calculation need not be performed is inconsistent with the guidance regarding the assurance of the operability of containment isolation valves. One of two SBO response valves in the steam tunnel identified by the licensee, i.e. the HPCI Steam Supply (MO 2035), is a containment isolation valve. The licensee made the assumption that this valve would only need to operate, (to open), upon the initiation of HPCI, which is in the beginning of the SBO event. As stated below (item 5, Containment Isolation), the licensee needs to ensure that all containment isolation valves can be fully closed, if needed, to provide containment integrity. By not performing the temperature calculation, the operability of containment isolation valve MO 2035 has not been demonstrated. The licensee needs to verify that this valve remains operable to isolate containment if needed during an SBO event.

5. Containment Isolation

Licensee's Submittal

The licensee stated that the plant list of containment isolation valves (CIVs) was reviewed to verify that CIVs which must be operated under SBO conditions can be positioned, with indication, independent of the preferred and blacked-out Class 1E AC power supplies. The licensee concluded that no modifications or procedure changes are necessary to ensure containmont integrity under SBO conditions.

The licensee added (13) that additional exclusion criterion was considered for CIVs which are on lines that terminate below the surface of the suppression pool based on the NRC exemption granted for testing these valves for leak tightness under 10 CFR 50 Appendix J. The licensee claimed that since the suppression pool provides an effective water seal against the release of fission products, valves located in these lines are not relied upon to perform a containment isolation function. Therefore, these valves are not required to operate or be capable of being closed for the purpose of establishing appropriate containment integrity.

Review of Licensee's Submittal

The list of containment isolation valves in the USAR was reviewed to determine which valves could not be excluded by the five criteria given in RG 1.155 and NUMARC 87-00. Those that could not be excluded were considered "valves of concern." An attempt was made to determine if these valves could be closed, with position indication, independent of the preferred and Class IE power supplies. The information available in the USAR does not allow complete verification of this capability.

The list of valves of concern developed by this evaluation was compared to the list supplied (16) by the licensee. Our review identified a number of valves, listed below, that appear to be valves of concern, but do not appear on the licensee's list in reference 16.

Penetration	Description	Valves	Size
X 8	Primary Steam Drain	MO2373/MO2374	3"
X 10	Steam to RCIC	MO2075/MO2076	3"
X 11	Steam to HPCI	MO2034/MO2035	8"
X 12	RHR Supply	MO2029/2030	18"
X I4	RWCU Supply	MO2397/MO2398	4"
X I6A	Core Spray	MO1752/MO1754	8"
X 16B	Core Spray	MO1751/MO1753	8"
X 17	Head Cooling	MO2027/MO2026	4"
X 224A	RHR B Suction	MO1987	20"
X 224B	RHR A Suction	MO1986	20"
X 225	HPCI Suction	MO2061/MO2062	16"
X 226A/B	Core Spray Suction	MO1742/MO1741	12"
X 227	RCIC Suction	MO2100	6"

We recognize that some of these valves are DC-operated, however we do not have information to confirm which ones could be closed independent of AC power sources.

The additional criterion assumed by the licensee in reference 13 to exclude CIVs that have a water seal is not among the exclusions criteria allowed by NUMARC 87-00 or RG 1.155. Although these valves have received exemption from leak rate testing under 10 CFR 50 Appendix J, this cannot be construed as an exclusion criterion from closure capability requirements. For this reason, those valves excluded under this provision need to be treated as other CIVs not excluded by the five criteria allowed in RG 1.155.

The assurance of the containment integrity requires that the operators be aware of CIVs positions at all time. Since during an SBO the AC-operated valves will not have position indications in the control room, the licensee needs to list in an appropriate procedure all CIVs that cannot be excluded by the five criteria given in RG 1.155 and are either normally closed or open and fail as-is upon loss of AC, and identify the actions necessary to ensure that these valves are fully closed, if needed. The valve closure needs to be confirmed by position indication (remote, local, mochanical, etc.).

6. Reactor Coolant Inventory

Licensee's Submittal

The licensee stated that the ability to maintain adequate reactor coolant system (RCS) inventory to ensure that the core is cooled has been assessed for four hours. A plant-specific analysis based on the generic analyses in NUMARC 87-00 was used for this assessment. The expected rates of reactor coolant inventory loss under SBO conditions do not result in more than a momentary core uncovery. Therefore, RCS make-up systems under SBO conditions are not required to maintain core cooling under natural circulation (including reflux boiling).

Review of Licensee's Submittal

Monticello has two recirculation pumps. The licensee assumed (reference 15, item 2) that each would leak 70 gpm through its seal which, when combined with the maximum allowed Technical Specifications leakage of 25 gpm, gives a total RCS leak rate of 165 gpm during the SBO event. When combined with the make-up required to remove decay heat, approximately 76,540 gallons will be needed to replenish the primary system inventory over a four hour SBO. The steam powered HPCI pump has sufficient capacity to supply the needed make-up, and the condensate storage tank and suppression pool contain sufficient volume to support the needed make-up for both decay heat removal and primary systems losses.

Note: The 18-gpm recirculation pump seal leak rate was agreed to between NUMARC and the NRC staff pending resolution of Generic Issue (GI) 23. If the final resolution of GI-23 defines higher recirculation pump seal leak rates than assumed for the RCS inventory evaluation, the licensee needs to be aware of the potential impact of this resolution on its analyses and actions addressing conformance to the SBO rule.

3.3 Proposed Procedures and Training

Licensee's Submittal

The licensee stated (13) that plant procedures have been reviewed and modified as required to meet the guidelines in NUMARC 87-00, Section 4, in the following areas:

1. Station Blackout response guidelines, (Procedure C.4-B.9.02.A)

2. Severe weather (Plant Inspection Program, Acts of Nature A-6), and

3. AC power restoration (Procedure C.4-B.9.02.A).

Further, the licensee stated (13) that the SBO response plant procedure has been reviewed and procedure changes will be implemented no later than the second refueling after notification is provided by the NRC staff.

Review of Licensee's Submittal

We neither received nor reviewed the affected procedures. These procedures are plant-specific actions concerning the required activities to cope with an SBO event. The licensee identified the procedures that have been or need to be modified and/or created to cope with an SBO event. It is the licensee's responsibility to revise and implement these procedures, as needed, to mitigate an SBO event and to assure that these procedures are complete and correct, and that the associated training needs are carried out accordingly.

3.4 Proposed Modifications

Licensee's Submittal

The licensee stated (12 and 13) that there are no plant modifications required to attain the proposed coping duration of four hours, however one modification was required to cope with a four-hour SBO. The proposed modification is to transfer the power supply for the alternate safety relief valve nitrogen supply to the Class 1E inverters. An additional modification was identified in reference 16, which requires the transfer of the power supply for temperature recorder TR 23-115 to panel Y-80 which is supplied from the Division 2 Class 1E inverter.

Review of Licensee's Submittal

This review identified no additional modifications needed to cope with a four-hour SBO event. The licensee-identified modifications are consistent with the guidance provided by NUMARC 87-00 and RG 1.155.

3.5 Quality Assurance and Technical Specifications

The licensee did not provide documentation on how the plant complies with the requirement of RG 1.155, Appendices A and B.

4.0 CONCLUSIONS

Based on our review of the licensee's submittals and the related supporting documents, we find that Monticello's submittal conforms to the requirements of the SBO rule and the guidance of R.G. I.155 with the following exceptions:

1. **Class 1E Battery Capacity**

   The capability of the Class 1E station batteries to support SBO loads for four hours needs to be verified considering an aging correction factor of 1.25 for all batteries and a minimum cell voltage of 1.80 and 1.82 VDC for Division I and II 250 volt battery, respectively. Further, the licensee needs to confirm that the emergency lighting is adequate to support operations under SBO conditions in order to justify shedding the emergency lighting loads from the Class 1E batteries.

2. **Loss of Ventilation**

   The licensee needs to revise the control room temperature rise calculation using a more conservative heat generation rate for occupants and possibly increasing the initial temperature in the control room, depending on the existence of a technical specification on the operability of the control room's HVAC system.

   Also, a temperature calculation in the steam tunnel and an evaluation of the reasonable assurance of operability of the containment isolation valves in the steam tunnel are needed.

3. **Containment Isolation**

   The assurance of the containment integrity requires that the operators be aware of CIVs positions at all time. Since during an SBO the AC-operated valves will not have position indications in the control room, the licensee needs to list in an appropriate procedure all CIVs that cannot be excluded by the five criteria given in RG 1.155 and are either normally closed or open and fail as-is upon loss of AC, and identify the actions necessary to ensure that these valves are fully closed, if needed. The valve closure needs to be confirmed by position indication (remote, local, mechanical, etc.).

4. Quality Assurance and Technical Specifications

The licensee did not provide documentation on how the plant complies with the requirement of RG 1.155, Appendices A and B.

5.0 REFERENCES

1. The Office of Federal Register, "Code of Federal Regulations Title 10 Part 50.63," 10 CFR 50.63, January 1, 19B9.

2. U.S. Nuclear Regulatory Commission, "Evaluation of Station 8lackout Accidents at Nuclear Power Plants - Technical Findings Related To Unresolved Safety Issue A-44," NUREG-1032, 8aranowsky, P. W., June 1988.

3. U.S. Nuclear Regulatory Commission, "Collection and Evaluation of Complete and Partial Losses of Offsite Power at Nuclear Power Plants," NUREG/CR-3992, February 19B5.

4. U.S. Nuclear Regulatory Commission, "Reliability of Emergency AC Power System at Nuclear Power Plants," NUREG/CR-2989, July 1983.

5. U.S. Nuclear Regulatory Commission, "Emergency Diesel Generator Operating Experience, 1981-1983," NUREG/CR-4347, December 1985.

6. U.S. Nuclear Regulatory Commission, "Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44)," NUREG/CR-3226, May 1983.

7. U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, "Regulatory Guide 1.155 Station Blackout," August 1988.

B. Nuclear Management and Resources Council, Inc., "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC 87-00, November 1987.

9. Thadani, A.C., letter to W. H. Rasin of NUMARC, "Approval of NUMARC Documents on Station Blackout (TAC-40577)," October 7, 1988.

10. Thadani, A.C., letter with attachment to A. Marion of NUMARC, "Publicly Noticed Meeting, December 27, 1989," dated January 3, 1990 (confirming "NUMARC 87-00 Supplemental Questions/Answers," December 27, 1987).

11. Nuclear Safety Analysis Center, "The Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants," NSAC-108, Wyckoff, H., September 1986.

12. Musolf, D., letter to the Director of Nuclear Reactor Regulation, USNRC, "Loss of All Alternating Current Power Information Required by 10 CFR Part 50, Section 50.63 (c)(1)," dated April 17, 1989.

13. Parker, T. M., letter to the Director of Nuclear Reactor Regulation, USNRC, "Loss of All Alternating Current Power Information Required by 10 CFR Part 50, Section 50.63 (c)(1)," dated October 17, 1989.

14. Parker, T. M., letter to the Director of Nuclear Reactor Regulation, USNRC, "Supplemental Response to Loss of All Alternating Current Power Information Required by 10 CFR Part 50, Section 50.63 (c)(1)," dated March 29, 1990.

15. Memorandum from C. Monteith (NSP) to W. Long (NRC), "Monticello Station Blackout Submittal Questions/Answers," dated January 24, 1991.

16. Memorandum from C. Monteith (NSP) to W. Long (NRC), "Monticello Station Blackout Submittal Questions/Answers," dated February 15, 1991.

17. Monticello Updated Safety Analysis Report.

18. General Electric Licensee Topical Report, "Recirculation Pump Shift Seal Leakage Analysis," NEDO-24083, November 1978.

19. NRC Inspection Report 50-263/90018 dated December 14, 1990.

20. Parker, T. M., letter to the director of Nuclear Reactor Regulation, USNRC, "Status Report on Meeting the Requirements of 10 CFR 50.63(c)(1)," dated June 16, 1989.

21. ASHRAE Handbook, 1977 Fundamentals, Published by the American Society of Heating, Refrigerating and Air-Conditioning Engineering, Inc. 1977-Edition.

DISTRIBUTION

Docket File
NRC & LPDRs
PD31 Rdg File
BBoger
JZwolinski
LMarsh
PShuttleworth
AMasciantonio
OGC
EJordan MNBB-3701
AToalston 7/E/4
ACRS(10)
Monticello Plant File
BClayton, RIII

tation which you may propose as a result of this evaluation are adequate to meet the SBO rule. The staff is developing guidance for this follow-up inspection to verify the following:

a. Hardware and procedural modifications;

b. SBO procedures in accordance with RG 1.155, Position 3.4, and NUMARC 87-00, Section 4;

c. Operator staffing and training to follow the identified actions in the procedures;

d. EDG reliability program meets, as a minimum, the guidelines of RG 1.155;

e. Equipment and components required to cope with an SBO are incorporated in a QA program that meets the guidance of RG 1.155, Appendix A; and

f. Actions taken pertaining to the specific recommendations noted in the SE.

The guidance provided on Technical Specifications (TS) for an SBO states that the TS should be consistent with the Interim Commission Policy Statement on Technical Specifications. The staff has taken the position that TS are required for SBO response equipment. However, the question of how TS for the SBO equipment will be applied is currently being considered generically by the NRC in the context of the Technical Specification Improvement Program and remains an open item at this time. In the interim, the staff expects plant procedures to reflect appropriate testing and surveillance requirements to ensure the operability of the necessary SBO equipment. If the staff later determines that TS regarding the SBO equipment are warranted, you will be notified of the implementation requirements.

If you have any questions concerning this action please contact me at (301) 492-1337.

This requirement affects fewer than ten respondents, therefore, is not subject to Office of Management & Budget under P.L. 96-511.

Sincerely,

Original signed by W.O. Long for

Armand Masciantonio, Project Manager
Project Directorate III-1
Division of Reactor Projects, III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

*MONTICELLO BLACKOUT LETTER

LA/PD31	PM/PD31	D/PD31
PShuttleworth	AMasciantonio	TMarsh
8/22/91	8/22/91	8/24/91