H. L. Sumner, Jr.
Vice President
Hatch Project

Southern Nuclear
Operating Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7279

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Edwin I. Hatch Nuclear Plant
Response to Request for Additional Information on the DC systems Technical
Specifications Revision

Ladies and Gentlemen:

On February 8, 2005, NRC electronically provided SNC with a request for additional information concerning the July 20, 2004 Plant Hatch Technical Specifications revision request which proposes to implement the guidance of TSTF- 360 to the Hatch DC systems Technical Specifications.

You will find our responses in the enclosure. A transcription of the NRC question precedes the SNC response.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/OCV/sdl

Enclosure:

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

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Enclosure

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NRC Question:

General Design Criterion 17, "Electric power systems," of Appendix A, "General Design Criteria for Nuclear Power Plants,", to Title 10 of the Code of Federal Regulations (10 CFR) Part 50 requires, in part, that the safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Additionally, 10 CFR 50.63, "Loss of Alternating Current [AC] Power", requires, in part, that each light water cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout (SBO) as defined in 10 CFR 50.2. The specified duration shall be based on the following factors:

(i) The redundancy of the onsite emergency AC power sources;

(ii) The reliability of the onsite emergency AC power sources;

(iii) The expected frequency of the loss of offsite power, and;

(iv) The probable time needed to restore offsite power.

The reactor core and associated coolant, control, and protection systems, including the station service batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a SBO for the specified duration.

The initial conditions of design basis accidents and transient analyses in the Hatch Final Safety Analysis Report (FSAR) SAFETY ANALYSES Chapter 5, 6 and 14, assume engineered safety feature (ESF) systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system, and containment design limits are not exceeded.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analysis and is based upon meeting the design basis of the unit. This includes maintaining the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power sources or all onsite AC power sources; and *b. A postulated worst case single failure*

Hatch FSAR chapter 8, Section 8.4.2.2, "SBO Coping Analysis Assumptions," states that equipment needed to cope for the SBO coping duration is available at the site.

Provide a comparison of Division I and II components to show any key safety features that would be affected due to a loss of offsite power with one system having an inoperable battery. Will the loss of a Division I or II station service battery (A or B) alone result in the loss of any redundant features? What effect, if any, does this have on Hatch's SBO analyses (e.g. Coping capability)? Identify any procedures, compensatory measures, and/or analyses that may be associated with the above request.

Hatch Response:

With respect to station service batteries, Southern Nuclear is requesting a change to the Technical Specifications (TS) Completion Time from 2 hours to 12 hours. Regarding the General Design Criteria, 10 CFR 50.63, and Final Safety Analysis Report (FSAR) safety analyses, none of the assumptions or methodologies used in the safety analyses or assumed system availabilities are changing with respect to OPERABLE equipment. When the station service batteries are OPERABLE, all the current analyses for Anticipated Operational Occurrences (AOOs), Accidents, or Special Events such as Station Blackout (SBO) remain the same. SNC is not requesting any changes in the way Plant Hatch complies with existing regulations and approved analyses for events and accidents. However, when safety related equipment is inoperable, the TS recognize that the plant is in a degraded condition. The degraded condition may be characterized by not being able to meet all assumptions of a particular FSAR safety or event analysis. The equipment needed to cope with SBO and other design and licensing basis events is indeed available at the site, but may not be available when the plant is in a degraded condition such as one requiring entry into a TS Action Statement. In such a case, the TS will allow operation only for a limited period of time. If, in that limited period of time the condition cannot be corrected, the TS require a plant shutdown, or in some cases other compensatory actions. For an inoperable station service battery, the TS currently allow operation for 2 hours prior to requiring a unit shutdown. SNC requests that time be increased to 12 hours, which is still a very restrictive Completion Time.

The station service DC system consists of two divisions, each with one 125/250 V DC battery and three battery chargers (two chargers are normally in-service and one in standby) as described in the original submittal. The DC subsystems provide power to many loads, both safety and non-safety related. The station service DC system is designed such that loss of one division will not prevent the successful mitigation of design basis events, such as a Loss of Coolant Accident with a concurrent Loss of Offsite Power (LOCA/LOSP).

Some of the more significant loads and the potential consequences of their loss are described below:

The division I DC system provides power to the division I logic for the Residual Heat Removal (RHR) System and the Core Spray (CS) system, the two safety related Hatch low pressure emergency core cooling systems (ECCS). The logic provides automatic initiations for these systems on low reactor water level and high drywell pressure, the traditional LOCA signals. Division II DC supplies power to the division II RHR and CS logic systems. There are two divisions of RHR, each with two low pressure pumps, and two divisions of CS, each with one low pressure pump. The division logic crosses over, in other words, an initiation signal received in division I logic will crossover into division II logic, initiating the division II pumps as well as the division I pumps. Of course, an initiation signal received on division II would do the same thing. Therefore, a loss of the division I battery, for example, together with a design basis LOCA/LOSP event would still result in the initiation of both divisions of RHR and CS. Nevertheless, it is worth noting that one division of RHR and CS is capable of providing adequate core cooling. Therefore, a failure of one ECCS division to start would not result in unacceptable consequences for the LOCA.

Control power for the RHR and CS pumps is supplied by the diesel generator DC system, not the station service DC. As described in the July 20 submittal, the diesel generator DC system includes a total of five batteries, one for each diesel generator. The existing TS Limiting Condition of Operation (LCO) for diesel generator batteries (LCO 3.8.4.B) has a Completion Time of 12 hours. Although section 3.8.4 of the TS is being revised by this request, the 12 hour Completion Time for the diesel generator batteries is not being changed.

Additionally, the DC station service system supplies logic power to the Automatic Depressurization System (ADS). The ADS system consists of seven of the eleven safety/relief valves (SRV) and functions to automatically depressurize the vessel in a LOCA event. Its initiation signals include low reactor water level and high drywell pressure, although a low water level condition will initiate ADS without a high drywell pressure signal present provided the low level condition remains in effect after a predetermined time delay. Again, division I DC supplies division I ADS logic power and division II provides division II ADS logic power. Similar to the ECCS logic, either division I or division II DC logic power is capable of initiating the entire ADS system. Therefore, a loss of one station service battery would not jeopardize the automatic function of the seven ADS valves.

Furthermore, Hatch Unit 1 and 2 Emergency Operating Procedures (EOPs), 31EO-EOP-001, provide the operators with adequate procedural guidance to ensure that the available operating systems are used to mitigate the consequences of the LOCA/LOSP event. For example, the EOP directs the operator to use whatever systems are available to mitigate the event such as the High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), RHR, or CS systems. The EOPs also provide guidance to use other non-safety related injection systems should a particular situation degrade beyond design basis assumptions.

Also, the loss of one division of DC will not result in the loss of the containment cooling which may be necessary to ensure the integrity of the primary containment. For example, the LOCA scenario may necessitate drywell and/or wetwell sprays, which is accomplished via the RHR system. Since containment cooling is a longer term action for the LOCA, it is likely that the DC system battery chargers would have already been energized by the diesel generators. This notwithstanding, containment cooling is a manual operation. Therefore, loss of a division's auto-initiation logic for the RHR pumps will not affect spray operation. Furthermore, the drywell and suppression chamber spray valves are AC powered valves, and thus are unaffected by the loss of a division of station service DC.

A loss of one station service battery would, therefore, not prevent the ECCS systems from performing their core and containment cooling functions in the LOCA/LOSP event.

The station service DC system also supplies power to the DC valves of the RCIC system via division I DC, and to the HPCI system via division II DC. Neither of these systems are credited in the design basis large break LOCA analysis. However, they are credited in some AOOs, most notably loss of feedwater flow and SBO. In the loss of feedwater flow event, RCIC is the typical system assumed to recover level. However, HPCI could also be used in the event that RCIC is unavailable for any reason, such as a loss of DC. Additionally, the low pressure ECCS could also be used following a vessel depressurization to the shutoff head of the pumps. So there are ample options available for level recovery should a DC battery be out of service at the onset of a loss of feedwater event.

The RCIC system is also credited in the SBO analysis as the primary water source for the core (Re: Unit 2 FSAR Chapter 8.4). RCIC is credited because it is a low volume system (rated for approximately 420 gallons per minute) and a loss of coolant event is not assumed to occur with the SBO. In fact, no other event is assumed to occur with the SBO. Consequently, the high volume HPCI system (rated for approximately 4200 gallons per minute) is not necessary to mitigate the potential consequences of a SBO event. Nonetheless, should the division I DC battery, and thus RCIC, be unavailable the operators will use the HPCI system, which would be available for use since its valves are powered by division II DC. However, the HPCI system will not have its room coolers available due to the SBO, whereas analysis has shown that RCIC can be used for extended periods without room cooling. The SBO analysis assumes that both HPCI and RCIC will automatically initiate upon the loss of the feedwater pumps.

An abnormal operating procedure (AOP) was developed for the SBO event, 34AB-R22-003, "Station Blackout". Consistent with the analysis, the procedure assumes that both the HPCI and RCIC systems initiate on low water level due to the loss of feedwater flow. The procedure has the operator secure HPCI as soon as possible to preserve battery power, since its high volume is not needed. If the RCIC system is not available, the operator will continue to use HPCI to control level. The station blackout procedure will have the operator secure the HPCI system only if it is not necessary to recover water level.

The Hatch SBO coping duration is four hours. The 1B diesel is a swing diesel that can supply power to either the Unit 1 'F' emergency 4 kV AC bus or the Unit 2 'F' emergency 4 kV AC bus. Within one hour, the swing 1B diesel generator is assumed to be available (per the analysis) and therefore, either division's battery chargers could be placed in service. Once the 4 kV bus is energized, power can be supplied through a transformer to either the division I or division II 600 V AC bus, which would allow energization of either the division I or II station service battery charger. If the division I battery is out of service, rendering RCIC unavailable, the operator could elect to power the division II battery charger, thereby supplementing the DC power to the HPCI system and returning its room coolers to service. The 'F' 4 kV bus powers the 'C' and 'D' RHR pumps. Thus, the operator could also elect to place these pumps in service in either a core cooling, or containment cooling capacity. In all likelihood, due to the challenges placed on the primary containment as a result of the SBO, the RHR pumps will likely be used in the containment cooling modes along with the service water pump which also is supplied power by the 'F' bus. No CS pumps are powered from the 'F' bus.

The DC station service system also supplies the power necessary to manually open the SRVs. As previously mentioned, there are a total of eleven SRVs, seven of which are the ADS valves. The other four valves are designated as low/low set (LLS) valves. The LLS valves automatically control reactor pressure by opening and closing between two pressure setpoints. For example, the valves will open following a reactor scram at approximately 1050 psig and close at approximately 900 psig. The seven ADS valves have the capability to have their valve control power supplied by either division of DC. In fact, its design logic will automatically transfer the ADS valve control power to one division should the other division fail. The four LLS valves, however, do not have that feature. Control power for two LLS valves is supplied by division I DC and two by division II. Nevertheless, a loss of one division battery will not prevent the function of LLS since two valves will be available with the auto pressure control feature intact. Also, the loss of one division's battery will not prevent the operator from manually operating the SRVs, for example for pressure reduction to enable injection with the low pressure RHR pumps. The loss of one division of station service DC will only result in the loss of manual initiation capability for two of the eleven SRVs.

The SRVs primary purpose is to provide an overpressure protection function for the reactor pressure vessel. Accordingly, the eleven valves will lift at a reactor pressure of 1150 psig. This is a mechanical function which does not rely on electrical power. The SRV main disk will lift off its seat when sufficient differential steam pressure develops across the valve's pilot disk. The loss of a division of station service DC will therefore not affect this mechanical function.

Throughout the SBO event, the operator must, of course, remain mindful of the load on the DC busses. Accordingly, the SBO procedure provides guidance to remove loads that are not critically needed. The procedure provides both general guidance to be cautious with the battery loads, as well as providing specific loads that could possibly be removed.

Furthermore, a reactor vessel low level condition, indicated by water level at the scram initiation setpoint (0 inches referenced from instrument zero) will require entry into the previously referenced EOPs. The EOPs will guide the operator through all the available injection systems, including RCIC and HPCI, to recover core coolant inventory. This means that if the division I station service battery is out of service, rendering RCIC unavailable, plant abnormal and emergency operating procedures are in place to guide the operators to use of the HPCI system and any other system which may be available and necessary to recover water level. If the division II battery is unavailable, the HPCI system would be out of service and the operator would of course use the primary system, RCIC.

Finally, although not assumed in the analysis, the SBO procedure has provisions for the local manual operation of the RCIC system if no DC or AC power can be restored. The operator will run the system by local manual operation of the necessary valves. RCIC flow may be adjusted via manual operation of the RCIC turbine trip and throttle valve. The steps to perform this local and manual initiation of the system are explicitly written into an attachment to the SBO procedure.

Summarizing, should an SBO event occur with one station service battery being out of service, the situation would be degraded, but would most likely be successfully mitigated. Most significant would be a loss of the division I DC which would result in the loss of the primary water recovery system, RCIC. However, in that case, the HPCI system would be available to restore level. Also, the alternate power source (diesel generator 1B) is recovered within one hour, which would make it possible to supplement the DC power to the HPCI system via the battery chargers, and to return the room coolers to service.

NRC Question:

In Enclosure 4, page E4-15, of its July 20, 2004 submittal, the licensee stated:

"Due to the diverse PSA [probabilistic safety assessment] model input that the Station Service Batteries have it is conservatively stated that no planned maintenance will occur during the time that the proposed AOT [allowed outage time] is invoked. In addition the proposed Station Service Battery extended AOT will only be used for emergent Station Service Battery work."

The actions described above (i.e., no planned maintenance to be performed during the proposed extended Completion Time, and the proposed extended Completion Time will only be utilized for emergent work) are critical to the acceptability of the proposed changes. How will the licensee assure that these actions are accomplished?

Hatch Response:

The actions to ensure that the station service battery AOT will only be used for emergent work and that no planned maintenance will be performed during the proposed extended Completion Time will be proceduralized in 90AC-OAM-002-0, "Scheduling Maintenance."

Furthermore, the actions will be formally entered into the Plant Hatch commitment tracking system, which is governed by procedure 00AC-REG-002-0, "Commitment Identification and Tracking System". One of the purposes of the tracking system is to prevent the inadvertent deletion of a commitment without receiving the appropriate levels of review. While these actions do not represent an NRC commitment as characterized by NEI-99-04, Revision 1, "Guidelines for Managing NRC Commitments", the procedural actions to use the 12 hour Completion Time only for emergent work and to ensure that no planned maintenance is performed during the extended Completion Time will be identified in the commitment tracking system, as a commitment which originated in an NRC safety evaluation.

When a procedure is being revised, the procedure writer is required to verify that the change will not interfere with an existing commitment. If a commitment is involved, it cannot be changed without a review by the affected manager and the Performance Analysis Manager. In this case, the affected manager is the Hatch Licensing Manager.

Furthermore, the Maintenance Rule Scoping document, and other review efforts, have identified certain Systems, Structures, and Components (SSCs), including the station service batteries, as being potentially risk significant, dependent upon the plant operating mode. The previously mentioned "Scheduling Maintenance" procedure requires that a risk informed assessment be performed prior to removing these SSCs from service for the purpose of maintenance. This is a requirement of current plant procedures and represents an additional restriction above and beyond those committed to as part of this submittal. Even without the "no planned maintenance" commitment of this submittal, removal of a station service battery for planned maintenance would require an assessment that could, depending on other functions that were also out of service at the time, require contingency actions and compensatory measures to manage the risk. Depending on the determined level of risk, the proposed maintenance would require increasing levels of management approval.