DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

January 5, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: McGuire Nuclear Station
Docket Nos. 50-369, 50-370

Dear Mr. Denton:

Attached are responses to several concerns identified in the draft Safety Evaluation Report (SER) for the McGuire Standby Shutdown System. Please note that other concerns identified in the SER were answered in our letter of December 14, 1982.

Very truly yours,

Hal B. Tucker

REH:jfw
Attachment

cc: Senior Resident Inspector
McGuire Nuclear Station

Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

B021

NRC

A. The Licensee has designed the SSS to provide an alternate and independent means to achieve and maintain hot standby conditions for one or both of the McGuire units. A design basis for the SSS as stated in the Licensee's March 31, 1980 letter is that "destruction of all equipment and cabling within a single fire zone shall not preclude the capability to achieve and maintain hot standby". In a telephone conversation on January 18, 1982, the Licensee stated that, as used in the statement of the above design basis, "fire zone" is equivalent to fire area, a space enclosed within three hour fire rated barriers. As a consequence of the above-stated design basis, each fire area (except the containment buildings) has either a normal or a standby shutdown system train outside the area. The Licensee should document the above statements made over the telephone.
(Page 4 Draft SER)

Response

A. As used in the description of the SSS, a "fire zone" is equivalent to a fire area and refers to a space enclosed within three hour rated fire barriers. Each fire area (except the Reactor Containment Buildings) has either a normal or a standby shutdown system train outside the area.

NRC

B. In the event of a fire in a plant area which necessitates use of the SSF, certain operations are performed in the "A" switchgear room and the "B" division disconnect cubicle. These two areas are separated from other plant areas and from each other by three hour fire rated barriers. Access to the "B" division disconnect cubicle is from the "A" switchgear room. The information regarding the separation of the disconnect cubicle from the "A" switchgear cubicle and other plant areas was obtained by a telephone conversation of January 29, 1982. The Licensee should document the above information.
(Page 5 Draft SER)

Response

B. The "A" switchgear room and the "B" division disconnect cubicle are separated from other plant areas and from each other by three hour rated fire barriers.

NRC

C. "[The licensee should provide a description of means used to disconnect power, i.e., whether switches are used or if cable connectors must be separated. The licensee should describe the number of such operations to be performed in the "A" switchgear compartment and "B" switchgear disconnect cubicle. The licensee should state which categories of plant personnel will have the responsibility to perform these disconnect operations. The licensee should justify the quantity and types of manual disconnection operations that might be required in a fire emergency.]" (Page 6 Draft SER)

Response

C. Molded case breakers with Kirk-key interlocks are utilized to disconnect the normal Train A feeder and connect the alternate SSS feeder to each of the following items.

Motor Control Center	1EMXA-4	(600VAC)
Motor Control Center	2EMXA-4	(600VAC)
Motor Control Center	1EMXH-1	(600VAC)
Distribution Center	1EVDA-1	(125 VDC)
Distribution Center	2EVDA-1	(125 VDC)

A Train B disconnect enclosure is provided in the "A" switchgear room. There are three plug assemblies per unit to be disconnected to remove power from the Train B devices. The plug assemblies are Pyle National control connectors, catalog Nos. ZPLML-1820-676-PN (plug) and ZRLML-1820-676-SN (receptacle).

The Operations Group (Nuclear Equipment Operators) will have the responsibility to perform these disconnnect operations, in the event of transfer to SSF control.

NRC

D. The Licensee should verify that a coordination analysis has been performed for the breakers on the associates circuits and that the breaker coordination has been found acceptable.
(Page 8 Draft SER)

Response

D. As stated in our letter of October 21, 1981, the breaker coordination in question is provided as a normal goal of our system design. Coordination studies are performed as an integral part of our engineering process to insure that in all practical cases, a fault on a cable fed from a Class 1E bus will result in the feeder for the faulted cable being tripped before the feeder for a whole bus. These studies are performed using vendor curves for the protective devices used, i.e., breakers or fuses.

Where equipment limitations or other considerations result in a lack of guaranteed coordination between load protective devices and Class 1E bus incoming protective devices, an analysis is performed to insure the acceptability of the application. For example, this analysis insures that for a fire within a given area which could create a fault on a non safety cable fed from a "Division B" Bus, no SSF ("Division A") cables are routed within that fire area if coordination cannot be demonstrated.

These studies have been completed for McGuire and the results show a design in accordance with the above criteria.

NRC

G. The licensee should verify that valves connecting the discharge of the CVCS charging pumps to the safety injection flow path are operable from the control room. Alternatively, the licensee should verify that adequate time and manpower are available with the minimum shift crew, exclusive of fire brigade members, to operate these valves outside of the control room for all cases where the safety injection flow path would be used.
(Page 13 Draft SER)

Response

G. In order for flow from the centrifugal charging pumps to reach the reactor coolant loops by means of the safety injection piping the following combination of valves in the safety injection system (NI) must be opened.

1NI4A or 1NI5B and 1NI9A or 1NI10B

All four valves are located in the Auxiliary Building and can be controlled from the control room.

NRC

H. The Licensee should demonstrate that the standby makeup pump can quickly return reactor coolant level in the pressurizer to the normal shutdown range after a delayed isolation of the nuclear sampling lines, CVCS letdown lines and other leakage paths as discussed below. A delay time of 30 minutes should be used to evaluate leakage from these unisolated paths in accordance with draft ANSI Standard 58.8, ANSI N660, Revision 2, March 1981 which specifies an operator action time of 30 minutes for operations outside the control room.
(Page 16 Draft SER)

Response

H. The originally accepted design concept for SSS operation was based on a 10 minute capability to transfer control to the SSF. Hot short or spurious actuation within the first 10 minutes are not specifically part of the design basis. This is based on the extreme unlikelihood of multiple spurious operations resulting in unacceptable coolant loss essentially coincident with loss of multiple systems within a 10 minute period. Fires cannot instantaneously incapacitate all equipment in a large area but can only propagate in some real finite time to cause progressive degradation or loss of control of equipment. In the event of a fire in areas containing systems or components required for safe shutdown which cannot be extinguished promptly, operating personnel will be dispatched to the "A" switchgear room and the SSF where they will establish communication with the control room. As long as capability exists to perform vital reactor control and monitoring from the control room, the operator will maintain control from that location. If vital control and monitoring functions (e.g., reactor coolant pressure boundary, reactor coolant makeup capability) become unacceptably degraded or unavailable from the control room, a prompt transfer can be made and control established from the SSF. Utilization of this approach allows use of the preferred control location with prompt transfer to the SSF should control room controls and/or instrumentation be lost.

In the unlikely event that reactor coolant system inventory loss exceeds the capability of the standby makeup pump coincident with loss of remote control of the charging pumps, the centrifugal charging pumps can be started locally at the switchgear. By using the pull-out fuse blocks at the switchgear, the pumps can be started regardless of the status of control interlocks or potential damage to control cables.

NRC

I. Isolation Valves Inside Containment; Line Penetrants Containment

Letdown lines of this type have isolation valves inside containment that are part of the SSS. However, the Licensee has not discussed alternate isolation valves outside containment that could be used to isolate the flow path in the event of damage to cables for the SSS valves. The following letdown paths are in this category.

a. Normal reactor coolant pump seal water-return, and

b. CVCS excess letdown.

In the event of a fire outside containment, these lines could be isolated by manual operation of breakers in the "A" division switchgear room. However the reactor coolant system would lose inventory until the lines are isolated. The Licensee should include these unisolated flow paths in the evaluation requested for flow paths with redundant isolations valves discussed in Item 1 above. In the event of a fire outside containment, fire damage could prevent isolation of these lines and the Licensee has not discussed alternate isolation capability. The Licensee has stated that if these valves are opened due to a fire inside containment, the standby makeup pump would not be used. Normal CVCS charging pumps would be available for reactor coolant system makeup. The Licensee should discuss the method that would be used to isolate these flow paths for a fire inside containment and the time required to accomplish the isolation. The Licensee should discuss the adverse consequences of delayed isolation of these flow paths. (Including the availability of makeup water sources.)
(Page 16 Draft SER)

Response

A description of cable routes associated with these isolation valves, as well as available capabilities for performing alternate functions was submitted to the NRR on October 20, 1981. The referenced systems were noted as items 8 and 9 in that submittal.

These SSF isolation functions are initiated because of the low flow capabilities of the standby makeup pump. A fire inside containment will not effect equipment located outside containment where a much larger flow is available from the charging pumps. Therefore, these isolation functions are not necessary.

NRC

J. The Licensee should demonstrate thai the normal makeup pumps have adequate capability to maintain pressurizer level in the shutdown range with open and unisolated PORVs and head vent valves and with other open discharge flow paths discussed in Item 2 (Open Item I) above. The Licensee should also demonstrate that adequate makeup water supplies are available onsite to recover losses from the PORV's, head vent valves, and the discharge paths in Item 2 (Open Item I) above.
(Page 17 Draft SER)

Response

J. The scenario described is bounded by the LOCA analysis described in the FSAR.

NRC

K. Therefore, the Licensee should separate cables for the "A" division RHR isolation valve from cables to the "B" division RHR isolation valve in accordance with the guidelines of Appendix R in all fire areas inside and outside of containment.
(Page 20 Draft SER)

Response

K. Because the subject cables from the valves to the control room are armored, external shorts are not considered possible. For the "A" division valve, the control circuit conductors are routed in separate, armored cables such that any internal shorts will not cause the valve to open. Therefore, separating "A" and "B" division cables for the RHR isolation valves is not necessary.

NRC

M. The Licensee should demonstrate that a method would be available to supply the spent fuel pool with borated water after a fire in any fire area outside containment. The Licensee should present the results of an analysis to verify that makeup to the spent fuel pool can be initiated in time to maintain a water level of at least 10 feet above the active fuel.
(Page 22 Draft SER)

Response

M. Analyses verify that makeup to the fuel pool can be initiated in time to maintain at least 10 feet of water level above the spent fuel assemblies. Assuming a maximum draw down rate of 26 gpm and worst expected heat load, operators will have more than 2 days to take damage control measures to restore makeup capability.

With offsite power, makeup to the fuel pool uses water from the RWST or the RMWST. The RWST and RMWST are located at ground elevation 760'. The minimum RWST level, during plant operation per Tech Spec., is approximately elevation 780'. The full RMWST level is approximately elevation 783'. Spent fuel pool minimum water level is elevation 771' 4 3/4". Due to relative locations between required spent fuel pool level and RWST or RMWST water levels, makeup flow is by gravity and can be initiated by manual valve operation outside the fuel building.

NRC

O. The Licensee should verify that steam generator indication would be avilable in the pump room after postulated fires or verify that the pump room operator will be in constant communication with the SSF control room where dedicated steam generator level instrumentation is available. (Page 25-26 Draft SER)

Response

O. There will be a constant communication link between the SSF and AFW pump room consisting of sound powered phone circuits. Plant telephones could also be used for most scenarios. In addition, portable radios will be available for use in case the plant telephones and the sound-powered phones are out of service.

NRC

P. The Licensee should verify that for a fire in the motor driven AFW pump room, the AFW flow could be controlled from the main control room. If this cannot be verified, the Licensee should describe the method by which an operator at the AFW pump turbine steam supply valves could follow steam generator level.
(Page 26 Draft SER)

P. Response

The only local operation station which requires immediate access is the turbine driven auxiliary feedwater pump room. In order to reach this room, an operator would pass through the motor driven auxiliary feedwater pump room (the turbine driven and motor driven pump rooms are separated by a 3 hr. fire barrier). A fire in the motor driven pump room could possibly delay operator action to throttle feedwater to individual Steam Generators. However, an alternate method of Steam Generator level control can be accomplished by throttling the steam supply to the turbine driven pump utilizing a valve located in the dog house area which would be unaffected by a fire in the motor driven pump room. Communication with the SSF will be accomplished using portable radios if the plant telephones and the sound-powered phones are out of service.

NRC

Q. The Licensee should discuss the worst-case effect of a fire on the atmospheric dump valves and demonstrate that the SSS can maintain safe hot standby conditions in the event of such postulated damage. (Page 27 Draft SER)

Response

Q. The Main Steam Isolation Valve (MSIV) and Power Operated Relief Valve (PORV) associated with one steam generator are closed upon transfer to the standby shutdown system to assure a source of steam for the turbine driven auxiliary feedwater pump. Procedures will require that the remaining MSIV's and PORV's be closed before initiating the transfer of control to the SSF.

It is considered incredible for any one MSIV or PORV associated with one of the other three steam generators to spuriously open by virture of the depth of design. Each valve is pneumatically controlled and requires air pressure to keep the valve open. In the pneumatic control line of each MSIV there are two solenoids, any one of which will cause the respective MSIV to close when deenergized. In the pneumatic control line of each PORV there are three solenoids, any one of which will cause the respective PORV to close when deenergized. Shorts in any of these circuits are much more likely to be a short to ground resulting in fuse or breaker isolation (thus closing the valve) than the precise hot short needed to open a single solenoid. The occurrence of the precise hot shorts necessary to energize redundant solenoids is considered extremely unlikely.

NRC

S. The Licensee should verify that wires for this SSS instrument (RCS pressure indication) are separated by Appendix R distances/barriers from wires for at least one channel of normal reactor coolant system pressure instrumentation in all plant fire areas including containment. (Page 29 Draft SER)

Response

S. Cables for the SSS and normal plant reactor coolant system transmitters have been reviewed to verify compliance with Appendix R separation requirements. In areas outside the Reactor Building, cables required for or associated with the primary method of shutdown are physically separated by the equivalent of a three hour rated fire barrier from cables required for or associated with the redundant or alternate method of shutdown.

In the Reactor Building, spatial separation between the normal plant and SSS transmitters can be noted in the following chart:

SSF DEDICATED TRANSMITTER (NOTE 1) DEVICE #	NORMAL PLANT TRANSMITTER (NOTE 2) DEVICE #	CHANNEL	SEPARATION (ft) (NOTE 3) HORZ	VERT	FUNCTION
1NCPT 5121	1NCPT5150	2	56	NA	NC System
	1NCPT5160	1	23	NA	
	1NCPT5170	3	34	NA	
	1NCPT5171	4	116	NA	

NOTES:

1. SSF dedicated transmitters are located in the Annulus.
2. The normal plant transmitters are located in containment with cables passing through the Annulus.
3. These are the shortest distance between the SSF transmitter and the cable of the normal plant transmitter as it passes through the Annulus.

In the annulus for each unit, detection and automatic sprinklers are provided in accordance with Appendix R, Section G.2.a, as stated for Unit 2 in the October 12, 1982 letter to H. R. Denton from H. B. Tucker.

NRC

U. The Licensee should verify that wires for this SSS instrumentation (pressurizer level indication) are separated by Appendix R distances/ barriers from wires for at least one channel of normal pressurizer level instrumentation in all fire areas including containment.
(Page 30 Draft SER)

Response

U. Cables for the SSS and normal plant pressurizer level transmitters have been reviewed to verify compliance with Appendix R separation requirements. In areas outside the Reactor Building, cables required for or associated with the primary method of shutdown are physically separated by the equivalent of a three hour rated fire barrier from cables required for or associated with the redundant or alternate method of shutdown.

In the Reactor Building, spatial separation between the normal plant and SSS transmitters can be noted in the following chart:

SSF DEDICATED TRANSMITTER (NOTE 1) DEVICE #	NORMAL PLANT TRANSMITTER (NOTE 2) DEVICE #	CHANNEL	SEPARATION (ft) (NOTE 3) HORZ	VERT	FUNCTION
1NCLT5151	1NCLT5150	2	22	NA	Pressurizer
	1NCLT5160	1	136	NA	Level
	1NCLT5170	3	147	NA	

NOTES:

1. SSF dedicated transmitters are located in the Annulus.
2. The normal plant transmitters are located in containment with cables passing through the Annulus.
3. These are the shortest distance between the SSF transmitter and the cable of the normal plant transmitters as it passes through the Annulus.

In the annulus for each unit, detection and automatic sprinklers are provided in accordance with Appendix R, Section G.2.a, as stated for Unit 2 in the October 12, 1982 letter to H.R. Denton from H.B. Tucker.

NRC

V. The Licensee should verify that wires for this SSS Instrumentation and control circuitry are separated by Appendix R distances/barriers from wires for plant steam generator level indication and control such that level and automatic valve operation would be available independent of fire damage in any plant fire area including containment. (Page 31 Draft SER).

Response

V. Cables for the SSS and normal plant steam generator level transmitters have been reviewed to verify compliance with Appendix R separation requirements. In areas outrside the Reactor Building, cables required for or associated with the primary method of shutdown are physically separated by the equivalent of a three hour rated fire barrier from cables required for or associated with the redundant or alternate method of shutdown.

In the Reactor Building, spatial separation between the normal plant and SSS transmitters can be noted in the following chart:

V. Response (Cont.)

SSF DEDICATED TRANSMITTER (NOTE 1) DEVICE #	NORMAL PLANT TRANSMITTER (NOTE 2) DEVICE #	CHANNEL	SEPARATION (ft) (NOTE 3) HORZ	VERT	FUNCTION
1CFLT6080	1CFLT6000	1	8	18	Steam Generator
	1CFLT5510	2	54	NA	A Level
	1CFLT5500	3	NA	2	
	1CFLT5490	4	90	NA	
1CFLT6090	1CFLT5540	1	86	NA	Steam Generator
	1CFLT6010	2	20	NA	B Level
	1CFLT5530	3	75	NA	
	1CFLT5520	4	2	NA	
1CFLT6100	1CFLT5570	1	86	NA	Steam Generator
	1CFLT6020	2	36	NA	C Level
	1CFLT5560	3	91	NA	
	1CFLT5550	4	NA	13	
1CFLT6110	1CFLT6030	1	29	NA	Steam Generator
	1CFLT5600	2	156	NA	D Level
	1CFLT5590	3	20	NA	
	1CFLT5580	4	132	NA	

NOTES:

1. SSF dedicated transmitters are located in the Annulus.
2. The normal plant transmitters are located in containment with cables passing through the Annulus.
3. These are the shortest distance between the SSF transmitter and the cable of the normal plant transmitters as it passes through the Annulus.

In the annulus for each unit, detection and automatic sprinklers are provided in accordance with Appendix R, Section G.2.a, as stated for Unit 2 in the October 12, 1982 letter to H. R. Denton from H. B. Tucker.

NRC

W. Thus, the licensee should provide indication of auxiliary feedwater flow rate to each steam generator at the SSF control room or other SSS control station. Wires for this instrumentation should be separated by Appendix R distances/barriers from wires for normal auxiliary feedwater flow rate instrumentation so that the indication would be available independent of fire damage in any fire area including containment.
(Page 31-32 Draft SER)

Response

W. Since auxiliary feedwater flow rate only affects steam generator level, the key parameter to monitor is steam generator level. Dedicated instrumentation for steam generator level is provided in the standby shutdown facility. This instrumentation is powered from the dedicated SSS power supplies and the Appendix R circuit separation from normal plant indication has been verified.

Without offsite power, a pressure gage located on the turbine-driven AFW pump suction could be used for verification of flow. With offsite power available, local instrumentation would be available for flow to each steam generator and for pump discharge pressure. The flow indication loop is an electronic, battery-backed circuit and the discharge pressure indication loop is pneumatic.

NRC

X. The Licensee should verify that wires for this instrumentation (standby makeup pump discharge flow) are separated by Appendix R distances/ barriers from cables for at least one division of the normal charging system in all fire areas except containment.
(Page 32 Draft SER)

Response

X. The cables for the standby makeup pump flow indication are routed through fire areas different from cables for the normal charging flow.

NRC

Y. The Licensee should include the SSF power sources in the existing emergency power system Technical Specifications.
(Page 34 Draft SER)

Response

Y. Duke Power Company does not believe that the SSF power sources should be included in the Technical Specifications, because the SSF power sources are not nuclear safety-related and are not required to prevent or mitigate the consequences of any accident described in the FSAR. If necessary, however, Duke will work with the NRC Staff in developing reasonable Technical Specifications for the SSF power sources.

NRC

Z. After completion of the detailed safe shutdown procedures for a fire in any station fire area, the Licensee should verify that adequate manpower will be available to achieve and maintain safe hot shutdown conditions in both units. Shutdown personnel should not include fire brigade members. The evaluation should consider the minimum onsite shift crew size. (Page 36 Draft SER)

Response

Z. Adequate manpower will be available to achieve and maintain safe hot standby conditions for both units. This includes the minimum onsite shift crew not including fire brigade members.

For each unit affected by a fire such that transfer of control to the SSF would be necessary, a minimum of one individual with a Reactor Operator (RO) license and two individuals capable of performing the duties of a Nuclear Equipment Operator (NEO) would be needed in addition to the fire brigade members. The RO would be stationed at the SSF. One NEO would perform the operations at the switchgear rooms to transfer control of the unit to the SSF. One NEO would be needed to control steam generator level. After the transfer of control is made, one NEO would be available at all times to perform miscellaneous duties, as needed, such as locally operating plant equipment and monitoring plant parameters.

The actual shift crew size will vary depending on the status os the units and the requirements of Technical Specifications. However, at all times, adequate manpower will be available to (1) achieve and maintain hot standby conditions for the unit(s) affected by the postulated fire, (2) comply with Technical Specifications minimum shift crew size requirements for the unaffected unit, and (3) comply with Technical Specifications concerning fire brigade requirements.

Realistically, more than the minimum required number of individuals would probably be available during a fire, because the McGuire shift crew size is normally greater than the minimum. Also, for a fire of sufficient concern to involve using the SSF, the McGuire Emergency Plan would be activated and support personnel called in.