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NUCLEAR PRODUCTION DEPARTMENT

August 21, 1981

U S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
File 0260/0862
Transmittal of Proposed FSAR Changes and Responses to NRC Questions
AECM-81/311

References:

1. Power Systems Branch Question 40.67
2. Discussion Item: Classification of Fire Detection Systems per NFPA 72D (G. Harrison 8/12/81)
3. Reactor Systems Branch informal questions resulting from meetings held during the week of May 18, 1981.

In response to your request for additional information, Mississippi Power & Light Company is submitting the enclosed materials updating information pertaining to the above referenced items.

Portions of the attached information were informally requested by Mr. G. Harrison (reviewer in fire protection area in CEB) in discussions with members of our staff on August 12, 1981. The revised response to question 40.67 is submitted in accordance with directions provided in discussions held with Mr. R. Giardina (reviewer in PSB) on August 17, 1981. This information represents proposed changes to the Grand Gulf Nuclear Station Final Safety Analysis Report (FSAR).

Proposed FSAR changes will be incorporated into the next amendment to the FSAR. If you have any questions or require further information, please contact this office.

Yours truly,

L. F. Dale
Manager of Nuclear Services

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Attachments: (See Next Page)
AE2M1

Attachments: 1. Question & Response 40.67

2. Discussion Item: Classification of Fire Detection System per NFPA 72D

3. TMI Related Issues: II.B.1
II.K.1.23
II.K.3.18
II.K.3.25

4. Revised Question and Response 211.133

cc: Mr. N. L. Stampley
Mr. G. B. Taylor
Mr. R. B. McGehee
Mr. T. B. Conner

Mr. Victor Stello, Jr., Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

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040.67 (9.5.8) You state in subsection 9.5.8.3 that the standby and HPCS diesel generator exhaust pipes extend, respectively, approximately 3 feet and 5 feet above the diesel building roof (El. 172"-0"). Figures 9.5-21 and 9.5-22 show the arrangement of the diesel generators, auxiliary equipment and location of the air intakes and diesel exhaust pipes. Discuss the ability of the exposed inline diesel exhaust pipes to withstand tornado missiles and assurance that one tornado missile will not damage all diesel exhaust pipes.

RESPONSE

The response to this question is given in part in revised subsection 9.5.8.3. Additionally, the diesel exhaust piping will be shortened from the present height above the diesel building roof to a height of 1 foot 3 inches above the diesel building roof. As described in subsection 9.5.8.3, the 2 foot high, seismic Category I parapet around the diesel building shields the shortened exhaust piping from horizontal missiles. The trajectory that would allow damage to all diesel exhaust pipes from one tornado missile as stated in the above question has been eliminated because of the shielding offered by the diesel building parapet.

Discussion Item: Classification of Fire Detection Systems per NFPA 72D

Response: Those fire detecton systems which are used to actuate suppression systems in safe-shutdown-related areas are classified as Class A in accordance with the requirements of NFPA 72D except for the Diesel Generator Rooms and the outside area where the ESF transformers are located.

As pointed out in discussions August 12, 1981, the Diesel Building rooms are provided with independent ultraviolet flame detectors which are separate from the detectors used to actuate the water suppression system. Therefore, the indication of a fire in that area will be alarmed in the control room. The ESF transformers are located outside and are a significant distance from other safe shutdown related areas.

As concluded from the discussion on August 12, 1981, upgrading of the fire detection systems providing suppression initiation in the previously mentioned areas will not be required for Grand Gulf.

TMI Items (NUREG-0737)*

5.1 II.B.1 (RCS Vents) MP&L must verify the following:

1. Head vent is powered by emergency power

2. SRVs have positive position indication in control room (this is also part of II.D.3)

3. RHR HX's have vents

4. RHR HX's vents are operable from control room

5. RHR HX's vents are powered by emergency power

RESPONSE

(1) The RCS head vent valves are used as a backup method for venting the Reactor Pressure Vessel (RPV) and are powered by non-class IE power. As described in subsection 18.1.19 the primary method used to vent the RPV is through the SRV's. Additional venting of the RPV is also provided by the RCIC turbine during an accident.

(2) The Safety/Relief Valves are provided with positive indication in the control room as described in a proposed revision of FSAR subsection 18.1.24 which was provided in response to an ICSB concern in the MP&L letter, AECM-81/308, dated August 21, 1981.

*This concern was directed to MP&L informally in telephone conversations with the Reactor Systems Branch (RSB) held during July 1981. The appropriate information from the above responses will be incorporated into the next available FSAR amendment.

TMI Items (NUREG-0737)*

5.3 II.K.1.23 (RPV Level Instrumentation) - The current response is too general. We suggest a reference to NEDO-24708A (see the LaSalle response for this item.)

RESPONSE

The response to the above item is provided in revised subsection 18.1.29.4.

*This concern was directed to MP&L informally (in meetings with the Reactor Systems Branch (RSB) held the week of May 18, 1981). The above referenced FSAR revisions will be incorporated into the next available FSAR amendment. The above statement of the RSB concern and response is provided for information only and will not be incorporated into the FSAR.

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18.1.29.4 Reactor Vessel Level Instrumentation (II.K.1.23)

REQUIREMENT

Describe all uses and types of reactor vessel level indications for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status.

RESPONSE

The response to the above requirement was forwarded to the NRC in letter AECM-80/26, dated March 19, 1980, which responded to IE Bulletin 79-08. Below is the resonse to this particular item provided at that time.

Reactor vessel water level is continuously monitored by 7 indicators or recorders for normal, transient, and accident conditions. Those monitors used to provide automatic safety equipment initiation are arranged in a redundant array with two instruments in each of two or more independent electronic divisions. Thus, adequate information is provided to automatically initiate safety actions and provide the operator with assurance of the vessel water level at all times. A more detailed description of water level instrumentation used in BWR/6 plants amd applicable to GGNS is provided in NEDO-24708A.

TMI Items (NUREG-0737)*

5.4 II.K.3.18 (ADS Logic Mod) - MP&L needs to take a position on this item. We prefer bypass of the high drywell pressure signal or a (timer on low level which bypasses the high drywell pressure signal upon runout). If the timer option is chosen, analyses must be provided to justify the timer settings.

RESPONSE

The response to the above items was provided in a proposed revision to FSAR subsection 18.1.39.6, which was provided in response to an ICSB concern in the MP&L letter, AECM-81/308, dated August 21, 1981.

*This concern was directed to MP&L informally in telephone conversations with the Reactor Systems Branch (RSB) held during July 1981. The appropriate information from the above response will be incorporated into the next available FSAR amendment.

TMI Items (NUREG-0737)*

5.5 II.K.3.25 (Loss of Offsite Power to Pump Seals Cooling) - The response needs to be clarified. Is the pump seal cooling system on emergency power (automatically) after a loss of offsite power? If not, the response is not acceptable. The 70 gpm leak rate argument is not acceptable.

RESPONSE

The response to the above item is provided in revised subsection 18.1.30.10.

*This concern was directed to MP&L informally (in meetings with the Reactor Systems Branch (RSB) held the week of May 18, 1981). The above referenced FSAR revisions will be incorporated into the next available FSAR amendment. The above statement of the RSB concern and response is provided for information only and will not be incorporated into the FSAR.

REQUIREMENT

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current (AC) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

RESPONSE

Mississippi Power & Light Company has participated in a BWR Owner's Group evaluation of the effect of loss of alternating current power on recirculation pump seals and has determined that no change design is necessary as described below.

The reactor recirculation pumps at Grand Gulf are provided with a mechanical shaft seal assembly. Two seals are built into a cartridge to facilitate replacement. Each individual seal in the cartridge is designed to withstand pump design pressure so that one seal can adequately limit leakage in the event the other seal fails. The pump shaft passes through a breakdown bushing in the pump casing to reduce leakage to less than 70 gpm in the event of a gross failure of both shaft seals.

During normal operation, the two sets of seals share the work load of the assembly. The sealing sufaces form two cavities in which pressure is measured and transmitted to the Operator Control Console in the control room. Pressure in the first cavity normally reads about 1050 psig, slightly above reactor pressures, and pressure in the second cavity is normally about 525 psig. Seal purge flow is provided into the first seal cavity from the Control Rod Drive (CRD) System. The CRD flow provides cool, reactor grade water to minimize seal wear and prolong seal life. Seal purging flow goes from the first cavity through a breakdown pressure orifice into the second cavity. Flow from the second cavity drains into the drywell chemical waste sump. The CRD system provides 3-5 gpm to the first seal cavity. Approximately 1 gpm goes through the seal cartridge and the remainder flows around the pump shaft and bushing into the impeller cavity. Alarms are provided on the seal purge flow lines and seal leakoff lines to indicate seal failure. The combination of seal pressure, seal flow and leakoff alarms permit the operator to analyze seal failures.

The recirculation pump seal cavity requires forced cooling due to the heat of both the reactor water and friction generated by the sealing surfaces. Cooling is provided by the Component Cooling Water (CCW) System. CCW flows in a cooling jacket surrounding the seal assembly. Temperature elements in the pump seal cavity monitor seal water temperature. Temperatures are recorded in the control room and high temperature alarms are provided on the Operator Control Console.

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Three CCW pumps are provided, and CCW pump"B" is powered from a class IE ESF power supply. In the event of loss of offsite power, the emergency diesel generators power the ESF bus feeding CCW pump "B". Within 10 seconds of loss of offsite power, the automatic load shedding and sequencing system repowers CCW pump "B". With the loss of offsite power, the Plant Service Water (PSW) System is no longer able to provide cooling to the CCW heat exchangers and the Standby Service Water (SSW) System automatically assumes the cooling function. However, if a LOCA is also present, transfer of the SSW to the CCW heat exchangers is manually initiated by the operator.

As a result of our review of containment isolation design (See subsection 18.1.26, Containment Isolation Dependability), the CCW supply and return lines through the containment have been designated "beneficial" and do not receive an automatic isolation signal so that CCW flow may continue to the recirculation pumps on loss of offsite power and/or LOCA events.

In summary, the recirculation pump seal coolers at Grand Gulf are provided with a reliable source of cooling water which can continue to operate following loss of offsite power. In addition, the seals are provided diverse instruments and alarms which alert the operator to seal failure. Should a gross seal failure take place the operator can simply close the suction and discharge valves on the affected pump and stop the leak. Thus, we believe this combination of design features eliminates the possibility of any adverse safety effects resulting from loss of seal cooling due to loss of offsite power and therefore no modifications are required.

211.133 (15.0) Provide a listing of the transients and accidents in Chapter 15 for which operator action is required in order to mitigate the consequences. For corrective actions required prior to 20 minutes, provide justification.

RESPONSE

The design and protection basis for the few situations where operator action is involved is and has been the 10 minute period. Lapse times of 10 minutes for these situations is considered appropriate. The necessity and justification of the operator corrective actions are discussed below.

Design Basis Accident Events

All immediate short term DBA event safety functions are automatic as well as manual. For NSSS-ESF systems and equipment, immediate long term safety actions might involve operator action at the 10 minute mark (as previously allowed and approved by the NRC). Long term required NSSS-ESF action can obviously be met since they involve the same equipment as safety function systems, which involve operator action at the 10 minute mark.

Anticipated Operational Transient Events

For all anticipated operational transients cited in Chapter 15, no operator corrective action is required to prevent the plant from exceeding safety design basis limits.

Operator action is required and utilized in order to:

a) Maintain the plant in a steady state condition,

b) Initiate safe and orderly shutdown,

c) Maneuver the plant from a condition that would necessitate safety action or,

d) Reduce the impact on plant systems operation due to a single operator error or a single equipment malfunction.

In no case would the operator's action or non-action result in an unacceptable effect on the health and safety of the general public.

In summary the general rules utilized in BWR technology include the following:

DBA............no operator action prior to 10 minutes.

Transient...immediate operator action is allowed to preclude unwarranted shutdown, ESF operation, unnecessary operation, and other non-safety actions.

Immediate operator action for transients is justifiable since this is his normal operational assignment.

The 10-minute operator action for accidents has been justifiable since the safety actions required are limited and require simple control initiations.

To address the concern expressed at the September 24, 1980 meeting with the NRC, the following is a discussion of operator actions for transients. Isolation events generally require the operator to perform the most actions. A general list of operator actions subsequent to an isolation event consists of the following:

a) Observe that all rods have inserted.

b) Observe that relief valves have opened for reactor pressure control.

c) Monitor and maintain reactor water level at required level.

d) Depending on conditions, initiate normal operating procedures for cool-down.

e) Verify HPCS and RCIC operation if auto initiation occurred due to level 2 trip.

f) Cool down the reactor per standard procedure if a restart is not intended.

To address additional concerns expressed by the NRC for the 10 to 20 minute time frame, the only operator action assumed for the LOCA analysis in the 10 to 20 minute time frame is:

- DBA LOCA assumes the operator diverts partial ECCS core cooling to containment cooling. This is a conservative assumption in that flow into the core is being diverted.

- Main Steamline Break Outside Containment Analysis assumes the operator depressurizes at the 10 minute mark.