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August 7, 1981

NUCLEAR PRODUCTION DEPARTMENT

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

Attention: Mr. Robert L. Tedesco, Assistant Director

Dear Mr. Tedesco:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
File 0260/0862
Unresolved Safety Issues
AECM-81/290

On June 12, 1981, your Mr. Kniel issued a memorandum on Grand Gulf discussing unresolved safety issues. This memorandum was given to Mississippi Power & Light Company (MP&L) on an informal basis by the NRC Project Manager for Grand Gulf Nuclear Station. Following our review and our meeting with the NRC on July 1, 1981, MP&L has compiled a response to each of the twelve (12) issues delineated in the June 12 memorandum.

Please find attached the above noted responses. It is our determination that this action should close the immediate effort required on unresolved safety issues by MP&L.

Yours truly,

John L. Richardson for

L. F. Dale
Manager of Nuclear Services

RMS/SHH/JDR:lm
Attachment

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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A-1 Waterhammer

Waterhammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions such as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Since 1971 over 200 incidents involving waterhammer in pressurized and boiling water reactors have been reported. The waterhammers (or steam hammers) have involved steam generator feedrings and piping, the residual heat removal systems, emergency core cooling systems, and containment spray, service water, feedwater and steam lines.

Most of the damage reported has been relatively minior, involving pipe hangers and restraints; however, several waterhammer incidents have resulted in piping and valve damage. The most serious waterhammer events have occurred in the steam generator feedrings of pressurized water reactors. In no case has any waterhammer incident resulted in the release of radioactive material.

Although waterhammer can occur in any light water reactor and over 100 actual and probable events have been reported in boiling water reactors, none have caused major pipe failures in a boiling water reactors such as Grand Gulf and none have resulted in the offsite release of radioactivity. As noted above, the most severe waterhammers observed to date have been in steam generators. Since the boiling water reactor does not utilize a steam generator, these worst cases are eliminated. Furthermore, any waterhammer which may occur in feedwater or main steam piping will not impair the emergency core cooling system since all ECCS water enters the reactor vessel via five separate reactor vessel nozzles independent of the feedwater and main steam piping.

In order to protect the emergency core cooling system against the effects of waterhammer, each ECCS pump is provided with its own jockey pump which provides a continuous supply of water to the high points of the emergency core cooling system discharge piping. Further assurance for filled discharge piping is provided by pressure instrumentation at the piping high point. An alarm sounds in the main control room if the pressure falls below a predetermined setpoint indicating difficulty maintaining a filled discharge line. Should this occur, or if an instrument becomes inoperable, the required action is identified in the Technical Specifications. Information is also provied in Q&R 211.68, .71, .190, .195.

Grand Gulf has installed a system to preclude waterhammer from occurring in emergency core cooling system lines. This system consists of jockey pumps to keep the emergency core cooling system lines water-filled so that the emergency core cooling system pumps will not start pumping into voided lines and steam will not collect in the emergency core cooling system piping. To ensure that the emergency core cooling system lines remain water-filled, vents have been installed and a Technical Specification requirement to periodically vent air from the lines has been imposed.

With regard to additional protection against potential waterhammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to preclude water slugs in steam lines and vapor formation in water lines,

(3) use of snubbers and pipe hangers, and (4) use of vents and drains.

Nonetheless, in the unlikely event that a large pipe break did result from a severe water event, core cooling is assured by the emergency core cooling systems and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided.

In the event that potentially significant waterhammer scenarios are identified which have not explicitly been accounted for in the design and operation of Grand Gulf, corrective measures will be implemented at that time. The need for measures beyond those already implemented has not been identified.

A-9 Anticipated Transients Without Scram

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (intiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired, then an "anticipated transient without scram," or ATWS, would have occurred.

Grand Gulf has been required to provide recirculation pump trip in the event of a reactor trip and to provide additional operator training for recovery from anticipated transient without scram events. In addition, Grand Gulf has implemented emergency procedures and operator training to cope with potential anticipated transient without scram events.

Operator training and action as described, in conjunction with the automatic recirculation pump trip, significantly improves the capability of the facility to withstand a range of anticipated transient without scram events, such that operation of this facility presents no undue risk to the health and safety of the public while this matter is under review.

The NRC stated intent for both of the two proposed rules being prepared for publication is to require Alternative 3A for the Grand Gulf plant. Grand Gulf is currently developing Alternative 3A design modifications for incorporation during its planned 1984 fuel reload shutdown unless deemed unnecessary by analysis or reduction of requirements. The other proposed ATWS rule would require fewer modifications and Grand Gulf would also be capable of complying with this rule if it were to be promulgated by the NRC. Grand Gulf will have ATWS operator procedures and RPT in place upon initial criticality. FSAR Section 15.8 also addresses this subject.

A-11 Reactor Vessel Materials Toughness

Resistance to brittle fracture is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in a nuclear reactor pressure vessel, three considerations are important. First, fracture toughness increases with increasing temperature; second, fracture toughness decreases with increasing load rates; and third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, power reactors are operated within restrictions imposed by the Technical Specifications on the pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material over the life of the plant is accounted for in Technical Specification limitations.

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection against reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

To assure adequate safety margins, adjustment to the nil ductility transition temperature (NDTT) and the developmental method for pressure/temperature curves are specified in 10 CFR 50 Appendices G and H. The amount of adjustment to the operating curves is a function of reference temperature, RT_{NDT} which depends upon the fast Neutron ($>$ 1 Mev) fluence and copper and phosphorus content in the RPV material. For BWR/6's, the copper and phosphorus content of the material is closely controlled. Furthermore, high upper shelf toughness is specified and all values for belt line material were in excess of 75 ft-lbs. The fast neutron fluence is low with respect to other reactor types because of the additional moderator (water) in the annulus between the core shroud and the RPV. Therefore, the reactor pressure vessel material toughness (A-11) issue is of relatively low concern for BWR/6's.

In Grand Gulf's case, the reactor pressure vessel (RPV) limiting material in the core belt line contains 0.04% copper and 0.012% phosphorus. The initial RT_{NDT} is $0^{o}F$. Based on a predicted adjusted reference temperature as a function of fluence and copper and phosphorus content, the end-of-life RT_{NDT} is predicted to be $26^{o}F$. On this basis, the Grand Gulf RPV has adequate safety margin with respect to the requirements of 10 CFR 50 Appendices G and H.

In addition FSAR Section 5.3.1.5.1.1 and Q&R 251.1, .3, .4, .5, .7, and .8 address this subject.

A-17 Systems Interaction in Nuclear Power Plants

The licensing requirements and procedures used in the design address many different types of systems interaction. Current licensing requirements are founded on the defense-in-depth principle. Adherence

to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. These design provisions supplemented by the current review procedures of the Standard Review Plan (NUREG-75/087), which require interdisciplinary reviews and which account, to a large extent, for review of potential systems interactions, provide for an adequately safe situation with respect to such interactions. The quality assurance program which is followed during the design, construction, and operational phases for each plant is expected to provide added assurance against the potential for adverse systems interactions.

A study by Sandia used fault-tree methods to identify component failure combinations (cut-sets) that could result in loss of a safety function. The cut-sets were reduced to minimal combinations by incorporating six common or linking systems failures into the analysis. The results of the Phase I effort indicate that, within the scope of the study, only a few areas of review procedures need improvement regarding systems interaction. However, the level of detail needed to identify all examples of potential system interaction candidates observed in some operating plants are not within the Phase I scope of the Sandia Study.

The Systems Interaction Branch, formed in the Office of Nuclear Reactor Regulation in April 1980, has been studying state-of-the-art methods that can be used to predict systems interactions. The initial effort, supported by three laboratory contracts, is underway; a range of methods is being considered and tested for feasibility against a sample of some systems interaction candidates derived from Licensee Event Report evaluations.

It is expected that the development of systematic ways to identify and evaluate systems interactions will reduce the likelihood of common cause failures resulting in the loss of plant safety functions. However, the studies to date indicate that current review procedures and criteria supplemented by the application of post-TMI findings and risk studies provide reasonable assurance that the effects of potential systems interaction on plant safety will be within the effects on plant safety previously evaluated.

The project administrative procedures (Project Procedures Manual and the Project Engineering Procedures Manual) provide the required guidance for interface between MP&L, GE, Bechtel and vendors. Specifically, the Project Procedures Manual identified the division of responsibility between MP&L, GE (NSSS supplier), Allis Chalmer Power Systems (Turbine Generator supplier) and Bechtel. These responsibilities consist of establishing Design Criteria, Final Design, Design Review, Procurement, Installation and Testing Services, Start-Up Services and Safety Analysis Reports. The Project Procedures Manual also identified the Material Assignment Schedule which specifies procurement responsibilities between MP&L, Bechtel-Jobsite and Bechtel-Gaithersburg. The Project Engineering Procedures Manual identifies Bechtel's design interface requirements. These requirements control internal, external and interdiscipline design review processes which includes interface between the Bechtel Engineering Team and MP&L, GE, Allis Chalmers Power Systems, suppliers/subcontractors and consultants. These processes contain provisions with regard to communications, documentation and change control. The Project Engineering Team also interfaces with the

following Bechtel entities: Home Office Construction Department, Specialist Groups; e.g., the Chiefs of Engineering and their staffs, other divisions and Bechtel companies; e.g., GeoTech, Materials and Quality Services (M&QS).

In addition, the interface between Bechtel, General Electric, and Mississippi Power & Light is tracked by the Project control log. A control number from the log is assigned to any Q correspondence that requires action by the recipient. This control number enables the Q-item to be tracked and ensures a follow-up on any open item for which a response has not been received.

To assure that all discipline interactions have identified all potential hazards to safety related equipment, the Project has formed the Engineering Review Team (ERT). This team will review the as-built condition of the plant for potential adverse effects to safety related equipment. The team is made up of members of all disciplines and all reports are coordinated with the responsible disciplines.

The following safety issues are included in the review by the Engineering Review Team:

Non-seismic Category I over seismic Category 1
High Energy Line Break
Flooding
Jet Impingement

Power and control cables are separated into three independent electrical divisions--I, II, and III--each serving separate safety-related systems. Operation of either Divisions I and III or II and III can be completely lost without affecting safe shutdown capability. Operation of Division I only or operation of Division II only is sufficient to achieve safe shutdown. The operability of either Division I or II is ensured by fire protection measures taken to ensure that a single fire cannot disable both divisions. Separation criteria utilized during the installation of safety-related cables provide protection against disabling redundant safety-related equipment by a cable fire. The criteria used for separation of safety-related cable trays and conduits are based on Regulatory Guide 1.75. The intent is to prevent a possible fire in one safety-related cable tray from spreading into a safety-related cable tray of a redundant electrical division and to prevent a possible fire in a non-safety-related cable tray from spreading into any safety-related cable tray. For a discussion of the Power and Control Cable Fire Protection Analysis, see FSAR Appendix 9A, Section 7.1.

A-39 Safety Relief Valve Pool Dynamic Loads

BWR plants are equipped with relief valves that discharge into the wetwell. Upon relief valve actuation, the initial air column within the SRV discharge line is accelerated by the high pressure steam flow and expands as it is released into the pool as a high pressure air bubble. The high rate of air and steam injection flow in the pool followed by expansion and contraction of the bubble as it rises to the pool surface produces pressure oscillations on the pool boundary. This effect is referred to as the air-cleaning phenomenon.

Experience at several BWR plants with pressure suppression containments has shown that damage to certain wetwell internal structures can occur

during safety/relief valve (SRV) blowdowns as a result of air clearing and steam quenching vibration phenomena.

In addition to the boundary loads, e.g., containment structures, reactor pedestal, the air injection and subsequent bubble motion produces pressure waves and water movement within the pool that produce drag loads on components in the pool.

Following the air-clearing phase, pure steam is injected into the pool. Condensation oscillations occur during this time period. However, the amplitudes of these vibrations are relatively small at low pool temperatures. Continued blowdown into the pool will increase the pool temperature until a threshold temperature is reached. At this point, steam condensation becomes unstable. Vibrations and forces can increase by a factor of 10 or more if the SRV continues to blow down. This effect is referred to as the steam quenching vibration phenomenon. Current practice for BWR operating plants is to restrict the allowable operating temperature envelope via Technical Specifications such that the threshold temperature is not reached.

With respect to Mark III containments, acceptance criteria for quencher loads have been established. These criteria were conservatively established based on the data base available. One of these criteria requires the applicants to assume that, for the events involving multiple valve actuations, the bubbles from each SRV discharge reach their peak pressures simultaneously and then oscillate in phase. In early 1978, GE proposed an equipment reevaluation program, which considers a statistical approach to determine the effects of bubble phasing considerations.

Recently, GE issued a Part 21 notification related to consecutive actuation of multiple safety/relief valves and simultaneous load increases for BWR Mark III water pressure-suppression containments. This concern resulted from a recent study performed by GE of the primary system pressure response following an isolation event. The results showed that more than one safety/relief valve could be actuated consecutively, as a result of a reactor isolation event. This SRV load combination has not been considered in the design. GE has also indicated that this concern is generic to all BWR containments and, therefore, is included in the task action plan. Grand Gulf has implemented a low-low setpoint to preclue this problem.

The approach taken by the owners groups consists of a number of comprehensive experimental and anlytical programs to establish and justify the SRV related pool dynamic loads for BWR Mark I and II designs. In addition, prototypical in-plant testing is proposed to confirm Mark I, II, and III SRV loads.

With respect to drag loads on submerged structures for both SRV and LOCA events, a generic analytical model is under development by GE which will be used for all BWR designs. For loads induced by air clearing, separate analytical models are under development to describe the two different types of discharge nozzles of the relief valve discharge lines; a ramshead model and a quencher model. The ramshead is a "Tee" fitting, whereas the quencher is a multi-branch diffuser type of nozzle.

Plans to use a quencher device rather than a ramshead have been proposed for Mark II type plants. The type of quencher, however, will be

different from that currently proposed by Mark III. The quencher design has been developed and tested in Germany by KWU.

With regard to the concern over subsequent actuations of multiple SRVs, GE has proposed a low-low setpoint SRV control logic to replace their current design for plants with Mark III containment. This logic is to ensure that no more than one relief valve will reopen following primary system transients. Therefore, the current containment design criteria related to SRV loads can be maintained.

For Mark III containments, acceptance criteria have been issued for SRV with quencher device. Although it is believed that the loads criteria are conservative, in-plant tests for confirmation will be performed. Since all Mark III containments use quencher devices, the pool temperature limits will not be an area of concern on the basis of current Mark III design. This information is provided in response to NRC Question 21.7.

The generic review of Mark III SRV loads is being conducted on the GESSAR-II document. The final phase of NRC review of the Mark III SRV loads began in May 1981 when the NRC staff issued two questions to GE on the reduced SRV load magnitudes. An informal meeting was held with the NRC staff in early July where GE presented responses to the two questions with a favorable response from the NRC staff. Formal GE responses to the two questions are scheduled to be submitted to the NRC staff in August 1981 at which time the NRC staff is expected to accept the GESSAR-II SRV loads and load methodology. A generic NRC NUREG on Mark III SRV loads is expected before the end of 1981. This generic NUREG is expected to lose NRC Task A-39 for Mark III plants. In addition, FSAR Appendix 6D, Sections 3B.5, .6, .7, .9, 3BA and Q&R 211.68 also address Mark III containment loads.

<u>A-40 Seismic Design Criteria - Short-Term Program</u>

NRC regulations required that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and in regulatory guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken to assure that these plants do not present an undue risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants.

The seismic design basis and seismic design of Grand Gulf have been evaluated at the operating license stage. Seismic design review of Grand Gulf was conducted using current licensing criteria and requirements. EG&G Idaho Incorporated, under contract to NRC, has conducted confirmatory review and independent structural analyses of the structural design of the Grand Gulf containment and auxiliary buildings. Further, in March 1980, NRC and EG&G conducted a detailed audit of civil/structural design methods for safety related structures. Finally, NRC requested further information to complete the FSAR review. This

information was provided in responses to NRC Questions 130.1, .3, .4, .14, .15, .16, .17, .18, .19, .20, .21, .22, .23, .24, .26, .29, .33, .35, .38, .40 and .41.

A-43 Containment Emergency Sump Reliability

Following a postulated loss-of-coolant accident, i.e., a break in the reactor coolant system piping, the water flowing from the break would be collected in the suppression pool. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water may also be circulated through the containment spray system to remove heat and fission products from the drywell and wetwell atmosphere. Loss of the ability to draw water from the suppression pool could disable the emergency cooling and containment spray systems.

The concern addressed by this Task Action Plan for boiling water reactors is limited to the potential for degraded emergency core cooling system performance as a result of thermal insulation debris that may be blown into the suppression pool during a loss-of-coolant accident and cause blockage of the pump suction lines. A second concern, potential vortex formation, is not considered a serious concern for Mark III containment due to the large depth of the pool and the low approach velocities (less than 6.4 fps).

With regard to potential blockage of the intake lines, the likelihood of any insulation being drawn into an emergency core cooling system pump suction line is very small. The potential debris in the drywell could only be swept into the suppression pool via the horizontal vents. Any pieces reaching the pool would tend to settle on the bottom and would not be drawn into the pump suction since the suction center line is 10.6 feet above the pool bottom. In addition, boiling water reactor designs employ strainers on the suction sized with flow areas 200% larger than the suction piping.

A discussion of the hydraulic performance of the suction lines including the potential for thermal insulation blockage of the suction strainers is contained in FSAR Section 6.2.2.2 (specifically, Pages 6.2-47 through 6.2-47b).

A discussion of the potential for vortex formation at the ECCS pump suction piping inlet is provided in Q&R 211.199.

A-44 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by, at least, two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes an offsite alternating current power connection, a standby emergency diesel generator alternating current power supply, and direct current sources.

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all alternating current power, i.e., a loss of both the offsite and the emergency diesel generator alternating current power supplies. This issue arose because

of operating experience regarding the reliability of alternating current power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. During each of these loss-of-offsite power events, the onsite emergency alternating current power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In addition, there have been numerous reports of emergency diesel-generators failing to start and run in operating plants during periodic surveillance tests.

A loss of all alternating current power was not a design basis event for the Grand Gulf facility. Nonetheless, a combination of design, operating, and testing requirements that have been imposed on the applicant will assure that these units will have substantial resistance to a loss of all alternating current and that, event if a loss of all alternating current should occur, there is reasonable assurance that the core will be cooled. These are discussed below.

If offsite alternating current power (three independent lines) is lost, three diesel-generators and their associated distribution systems will deliver emergency power to safety-related equipment. The design, testing, surveillance, and maintenance provisions for the onsite emergency diesels is described in Section 8.3 of the FSAR. The requirements include preoperational testing to assure the reliability of the installed diesel-generators in accordance with our requirements discussed in this report. In addition, Grand Gulf has implemented a program for enhancement of diesel-generator reliability to better assure the long-term reliability of the diesel-generators.

If both offsite and onsite alternating current power are lost, boiling water reactors may use a combination of safety/relief valves and the reactor core isolation cooling system to remove core decay heat without reliance on alternating current power. These systems assure that adequate cooling can be maintained for at least two hours, which allows time for restoration of alternating current power from either offsite or onsite sources.

Based upon a preliminary evaluation the only power available for the operation of equipment, lighting, instrumentation, controls and valve operation are the station batteries (ESF batteries A & B). These batteries will supply power to critical equipment needed during the transient. In addition non-divisional batteries D&E and their inverters are available to supply 120 volt AC in the control building should the need arise.

On loss of instrument air, the ADS/SRV air system has 7.6 days of air available for ADS operation assuming 3 actuations of all ADS valves, or approximately 14 days of air available for relief operation assuming no actuation. With the loss of instrument air and one air receiver, the ADS/SRV air system has approximately 33 hours of air available for ADS operation of the valves associated with the lost receiver. Additionally, upon loss of instrument air the ADS/SRV air system has air available for 100 SRV actuations in 6 hours for low-low setpoint relief function.

The equipment which is available to control reactor vessel level, temperature and pressure are the auto depressurization system

safety/reliefs (ADS) and the Reactor Core Isolation Cooling System (RCIC). In addition, emergency lighting and critical instrumentation is powered from the station divisional batteries.

Certain operator actions are required during the course of this event to ensure that sufficient water supplies are available, that equipment is maintained below abnormal operating temperature limits, to ensure that the suppression pool is heated in a uniform manner and that sufficient air and electrical supplies are available. These required actions are identified in the sequence of events. In general, at the onset of the transient, the operator is assumed to take no action for 10 minutes at which time, he has accurately assessed the situation and began taking necessary actions to mitigate the transient.

Our evaluation will continue on this task and will assess any failure developments.

A-45 Shutdown Decay Heat Removal Requirements

Following a reactor shutdown, the radioactive decay of fission products continues to produce heat (decay heat) which must be removed from the primary system. The principal means for removing this heat in a boiling water reactor while at high pressure is via the steam lines to the turbine condenser. The condensate is normally returned to the reactor vessel by the feedwater system, however, the steam turbine-driven reactor core isolation cooling system is provided to maintain primary system inventory, if alternating current power is not available. When the system is at low pressure, the decay heat is removed by the residual heat removal systems. This "Unresolved Safety Issue" will evaluate the benefit of providing alternate means of decay heat removal which could substantially increase the plants' capability to handle a broader spectrum of transients and accidents. The study will consist of a generic system evaluation and will result in recommendations regarding the desirability of and possible design requirements for improvements in existing systems or an alternative decay heat removal method if the improvements or alternative can significantly reduce the overall risk to the public.

The Grand Gulf reactors have various methods for the removal of decay heat. As discussed above, the decay heat is normally rejected to the turbine condenser and returned to the vessel by either the feedwater system or the reactor core isolation cooling system (from the condensate storage tank). If the condenser is not available (e.g., loss of offsite power), heat can be removed via the safety/relief valves to the suppression pool. Also, the high pressure core spray system is provided if the reactor core isolation cooling system is not available. Both of these systems can supply fluid to the vessel from either the condensate storage tank or the suppression pool. If the reactor core isolation cooling and high pressure core spray are unavailable, the reactor system pressure can be reduced by the automatic depressurization system so that cooling by the residual heat removal can be initiated. When the condenser is not used, the heat rejected to the suppression pool is subsequently removed by the residual heat removal system.

The reactor core isolation cooling and high pressure core spray systems at Grand Gulf have improvements over comparable systems at older boiling water reactors. The reactor core isolation cooling system has been upgraded to safety-grade quality (now required for all boiling water

reactors), and the high pressure core spray is powered by its own dedicated diesel so it can operate with an assumed loss of all other sources of alternating current power. Also, the residual heat removal system contains three pumps; the flow capacity of any single pump (A or B) is sufficient to easily remove the decay heat.

The development of shutdown cooling requirements from this evaluation could result in a range of potential impacts on GGNS. Assuming the current BWR systems were found adequate under the criteria to be developed, no modifications to the GGNS design would be required. At the other extreme, significant modifications to existing decay heat removal systems, or the requirement to provide additional decay heat removal capability could result.

Following the TMI accident, the industry performed and documented extensive analyses of feedwater transients and small-break loss-of-coolant accidents which supported the acceptability of current designs. In addition, GE has defined plant modifications to increase the reliability of the decay heat removal system, and is currently working to implement those modifications.

Future analyses regarding this issue will be performed in response to the rulemaking for Degraded Core Cooling. FSAR Section 15.2.9 and Q&R 211.126 and 212.20 also address this subject.

A-46 Seismic Qualification of Equipment in Operating Plants

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commerical nuclear power program. Consequently, the margins of safety provided in existing equipment to resits seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this "Unresolved Safety Issue" is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shut down the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

Due to changes in seismic design criteria and methods during the course of the commercial nuclear power program, margins could vary considerably. Therefore, a seismic qualification reassessment of safety-related mechanical and electrical equipment is required to an explicit set of guidelines.

The Grand Gulf licensing commitment, as stated in FSAR Section 3.9 and 3.10, is that non-NSSS seismic Category I equipment meet IEEE 344-1971, as supplemented by Branch Technical Position EICSB-10 and that NSSS seismic Category I equipment meet IEEE 344-1971; in addition, NSSS motor and air actuators, including electrical appurtenances, must meet IEEE 344-1975, Paragraph 6.6.6.

In February 1979, Grand Gulf received Question 110.43 which requested seismic and hydrodynamic response spectra information in anticipation of the Seismic Qualification Review Team (SQRT) site visit. The initial response to the question was provided in FSAR Amendment 33, September 1979, and a final response was provided in Amendment 47, April 1981.

In a March 1981, response (AECM-81/89, March 6, 1981; AECM-81/98, March 16, 1981) to an NRC letter dated January 27, 1981, requesting information concerning equipment qualification for seismic and hydrodynamic loads at the Grand Gulf Nuclear Station, a review was performed in order to assess the adequacy of the existing qualification based on the NRC Seismic Qualification Review Team (SQRT) requirements. The information was also provided in FSAR Amendment 47, April 1981. These requirements include:

a. IEEE 344-1975

b. Regulatory Guide 1.92

c. Regulatory Guide 1.100

d. Standard Review Plan 3.9.2

e. Standard Review Plan 3.10

f. Grand Gulf unique building vibratory response to seismic and hydrodynamic loads

Prior to the start of the SQRT-evaluation, a list of nuclear safety-related equipment was developed. This list is depicted in FSAR Tables 3.10-3 and 5. Equipment qualification results were reviewed in order to assess the adequacy of equipment qualification in light of the SQRT requirements. The qualification information is summarized in FSAR subsection 3.10.2.1.4 and 3.10.2.2.5 and in FSAR Tables 3.10-4 and 6. The SQRT audit has been held and the results are forthcoming.

A-47 Safety Implications of Control Systems

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential for a single failure such as a loss of a power supply, short circuit, open circuit, or sensor failure to cause simultaneous malfunction of several control features. Such an occurrence would conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is for a postulated accident to cause control system failures which would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Although it is generally believed that such control system failures would not lead to serious events or results in conditions that safety systems cannot safely handle, in-depth studies have not been rigorously performed to verify this belief. The potential for an accident that would affect a particular control system, and effects of the control system failures, may differ from plant to plant.

Therefore, it is not possible to develop generic answers to these concerns, but rather plant-specific reviews are required.

The Grand Gulf control and safety systems have been designed with the goal of ensuring that control system failures (either single or multiple failures) will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any "anticipated operational occurrence" or "accident." This has been accomplished by either providing independence between safety and nonsafety systems or providing isolating devices between safety and nonsafety systems. These devices preclude the propagation of nonsafety system equipment faults such that operation of the safety system equipment is not impaired.

A wide range of bounding transients and accidents is presently analyzed to assure that the postulated events would be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that control system failures (single or multiple) will not defeat safety system action. Specifically, these reviews have included:

1. IE Bulletin 79-27

   A series of tables has been developed which lists GGNS power sources down to the fuse level, to include alarm indications, instruments and control devices on these power sources. Completion of the tables with primary and secondary effects from loss of the power sources is in progress. Design modifications will be made as necessary when the determined effects have an adverse impact on plant safety.

2. NRC letter dated April 16, 1981, "Control Systems Failures"

   To address item (1) of this letter (identification of control systems failures which could impact plant safety), phenomena which could occur to initiate or worsen a transient/accident were determined. An exhaustive study was then made to determine all control systems failures which could result in the phenomena.

   Identification of the power panel, MCC, LCC, bus, transformer, battery and/or inverter, as applicable for each control system identified in Item (1) was made. A rearrangement of this information showed control systems with common power sources and the effects of cascading power losses.

   A determination of control systems identified in Item (1) that receive input signals from common sensors was completed.

   An evaluation of the effects of the loss of a common sensor or power source on the analyses presented in FSAR Chapter 15 is now being conducted.

3. NRC letter dated April 16, 1981, "High Energy Line Breaks and Consequential Control Systems Failures," IE Notice 79-22

   A matrix is being developed which shows the effects, if any, of high energy line breaks on control systems. If interaction

is discovered, the impact of failure of the applicable system upon the GGNS safety analyses will be evaluated.

A specific subtask of this "Unresolved Safety Issue" will be to study the reactor overfill transient in boiling water reactors to determine the need for preventative and/or mitigating design measures to preclude or minimize the consequences of this transient. Several early boiling water reactors have experienced reactor vessel overfill transients with subsequent two-phase or liquid flow through the safety/relief valves. Following these early events, commercial-grade high-level trips (level 8) have been installed at most boiling water reactors (including Grand Gulf) to terminate flow from the appropriate systems. These high-level trips are single failure proof and periodic surveillance is required by the Technical Specifications. No overfilling events have occurred since the level 8 trips were installed. In addition BWR/6's have a high level scram that precludes this concern.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Following a loss-of-coolant accident is a light water reactor plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal-water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal, and radiolytic effects of postaccident environmental conditions on containment protective coating systems and electric cable insulation.

Regulation 10 CFR Section 50.44 requires that the combustible gas control system provided be capable of handling the hydrogen generated as a result of degradation of the emergency core cooling system such that the hydrogen release is five times the amount calculated in demonstrating compliance with 10 CFR Section 50.46 or the amount corresponding to reaction of the cladding to a depth of 0.00023 inch, whichever amount is greater.

The accident at TMI-2 on March 28, 1979, resulted in hydrogen generation well in excess of the amounts specified in 10 CFR Section 50.44 As a result of this knowledge it became apparent to NRC that specific design measures are needed for handling larger hydrogen releases, particularly for small, low-pressure containments.

Recognizing that a number of years may be required to complete this rulemaking proceeding, a set of short-term or interim actions relative to hydrogen control requirements was developed and implemented. These interim measures were described in an October 2, 1980 Federal Register notice. For plants with Mark III containments such as Grand Gulf, the proposed interim rule specified that either it must be demonstrated that the containment can withstand hydrogen burns or explosions or a detailed evaluation of possible hydrogen control measures must be performed and the selected measures installed.

Grand Gulf was requested to comply with these interim measures prior to fuel load. In submittals made to the NRC on April 9 and June 19, 1981, the evaluation of alternate hydrogen control measures was evaluated, a

Hydrogen Ignition System (HIS) was selected and detailed evaluations of containment pressure and temperature response were demonstrated.

The HIS, as further described in a letter to NRC dated June 19, 1981, consists of glow plug igniters distributed throughout the containment and drywell. The HIS is designed to ignite hydrogen at low concentrations, thereby maintaining the concentration of hydrogen below its detonable limit and preventing containment overpressure failure. Containment response to the burning of hydrogen has been analyzed using the CLASIX-3 computer code developed by Offshore Power Systems. An analysis of the ability of essential equipment to survive the hydrogen burn environment is underway; the anticipated completion date is December 1981. The HIS will be installed and fully operable by the December 31, 1981, Unit 1 fuel load date.

Significant additional work is underway to demonstrate that the containment pressure and temperature response calculations are adequate, that potential detonations do not consitute a threat to safety, and that essential equipment will survive hydrogen burns resulting from operation of the HIS.

In addition, Mark III owners have formed an owners group to evaluate hydrogen control measures for Mark III containments, and the applicant is actively involved in the ongoing evaluations of that owners group.