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RECEIVED NOV 30 1981 U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos: 50-458
and 50-459

Mr. William J. Cahill, Jr
Senior Vice President - River Bend Nuclear Group
Gulf States Utilities Company
Post Office Box 2951
Beaumont, Texas 77704

Dear Mr. Cahill:

Subject: Requests for Additional Information Regarding the Status of Unresolved Safety Issues

The Generic Issues Branch has identified a need for additional information regarding the status of Unresolved Safety Issues pertaining to River Bend. This informational request is provided as enclosure (1). Your response to enclosure (1) should be provided no later than June 1, 1982.

Enclosure (2) is the Generic Issues Branch SER contribution for a recent BWR plant, Grand Gulf. This enclosure is provided for your information and to assist you in your responses.

Sincerely,

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
As stated

cc: See next page

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| SURNAME | RPerch | ASchwencer | | | | | |
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NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Enclosure 1

## REQUEST FOR INFORMATION

The Atomic Safety and Licensing Appeal Board in ALAB-444 determined that the Safety Evaluation Report for each plant should contain an assessment of each significant unresolved generic safety question. It is the staff's view that the generic issues identified as "Unresolved Safety Issues" (NUREG-0606) are the substantive safety issues referred to by the Appeal Board. Accordingly, we are requesting that you provide us with a summary description of your relevant investigative programs and the interim measures you have devised for dealing with these issues pending the completion of the investigation, and what alternative courses of action might be available should the program not produce the envisaged result.

There are currently a total of 26 Unresolved Safety Issues discussed in NUREG-0606. We do not require information from you at this time for a number of the issues since a number of the issues do not apply to your type of reactor, or because a generic resolution has been issued. Issues which have been resolved have been or are being incorporated into the NRC licensing guidance and are addressed as a part of the normal review process. However, we do request the information noted above for each of the issues listed below:

1. Waterhammer (A-1)
2. Anticipated Transient Without Scram (A-9)
3. Reactor Vessel Materials Tcughness (A-11)
4. Systems Interaction in Nuclear Power Plants (A-17)
5. Safety Relief Valve Pool Dynamic Loads (A-39)
6. Seismic Design Criteria (A-40)
7. Containment Emergency Sump Reliability (A-43)
8. Station Blackout (A-44)
9. Shutdown Decay Heat Removal Requirements (A-45)
10. Seismic Qualification of Equipment in Operating Plants (A-46)
11. Safety Implications of Control Systems (A-17)
12. Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)

Enclosure 2

# APPENDIX C

## NUCLEAR REGULATORY COMMISSION (NRC) UNRESOLVED SAFETY ISSUES

### C.1 Unresolved Safety Issues

The NRC staff continuously evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors; research results; NRC staff and Advisory Committee on Reactor Safeguards (ACRS) safety reviews; and vendor, architect/engineer and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to assure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to assure safety, e.g., the derating of boiling water reactors as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue prior to making licensing decisions. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long term operation of plants already under construction or in operation.

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. Certain of these issues have been designated as "unresolved safety issues" (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978). However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer-term generic review is underway.

### C.2 ALAB-444 Requirements

These longer-term generic studies were the subject of a Decision by the Atomic Safety and Licensing Appeal Board of the Nuclear Regulatory Commission. The Decision was issued on November 23, 1977 (ALAB-444) in connection with the Appeal Board's consideration of the Gulf States Utility Company application for the River Bend Station, Unit Nos. 1 and 2.

"In short, the board (and the public as well) should be in a position to ascertain from the SER itself--without the need to resort to extrinsic documents--the staff's perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny. Once again, this assessment might well have a direct bearing upon the ability of the licensing board to make the safety findings required of it on the construction permit level even though the generic answer to the question remains in the offing. Among other things, the furnished information would likely shed light on such alternatively important considerations as whether: (1) the problem has already been resolved for the reactor under study; (2) there is a reasonable basis for concluding that a satisfactory solution will be obtained before the reactor is put in operation; or (3) the problem would have no safety implications until after several years of reactor operation and, should it not be resolved by then, alternative means will be available to insure that continued operation (if permitted at all) would not pose an undue risk to the public."

This appendix is specifically included to respond to the decision of the Atomic Safety and Licensing Appeal Board as enunciated in ALAB-444, and as applied to an operating license proceeding Virginia Electric and Power Company (North Anna Nuclear Power Station, Unit Nos 1 and 2), ALAB-491, NRC 245 (1978).

## C.3 "Unresolved Safety Issues"

In a related matter, as a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977 to include, among other things, a new Section 210 as follows:

"UNRESOLVED SAFETY ISSUES PLAN"

> "SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter."

> The Joint Explanatory Statement of the House-Senate Conference Committee for the Fiscal Year 1978 Appropriations Bill (Bill S.1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

"SECTION 3 - UNRESOLVED SAFETY ISSUES"

> "The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned."

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report, NUREG-0410, entitled "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," describing the NRC generic issues program. The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to assure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of over 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as "Unresolved Safety Issues" for reporting to the Congress. The NRC review included the development of proposals by the NRC Staff and review and final approval by the NRC Commissioners.

This review is described in a report NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress," dated January 1979. The report provides the following definition of an "Unresolved Safety Issue:"

> "An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be accepable over the lifetime of the plants it affects."

Further the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an "Unresolved Safety Issue" is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, seventeen "Unresolved Safety Issues" addressed by twenty-two tasks in the NRC program were identified. The issues are listed below. Progress on these issues was first discussed in the 1978 NRC Annual Report. The number(s) of the generic task(s) (e.g., A-1) in the NRC program addressing each issue is indicated in parentheses following the title.

"UNRESOLVED SAFETY ISSUES" (APPLICABLE TASK NOS.)

1. Waterhammer - (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System - (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity - (A-3, A-4, A-5)
4. BWR Mark I and Mark II Pressure Suppression Containments - (A-6, A-7, A-8, A-39)
5. Anticipated Transients Without Scram - (A-9)
6. BWR Nozzle Cracking - (A-10)
7. Reactor Vessel Materials Toughness - (A-11)
8. Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports - (A-12)

9. Systems Interaction in Nuclear Power Plants - (A-17)
10. Environmental Qualification of Safety-Related Electrical Equipment - (A-24)
11. Reactor Vessel Pressure Transient Protection - (A-26)
12. Residual Heat Removal Requirements - (A-31)
13. Control of Heavy Loads Near Spent Fuel - (A-36)
14. Seismic Design Criteria - (A-40)
15. Pipe Cracks at Boiling Water Reactors - (A-42)
16. Containment Emergency Sump Reliability - (A-43)
17. Station Blackout - (A-44)

In the view of the staff, the "Unresolved Safety Issues" listed above are the substantive safety issues referred to by the Appeal Board in ALAB-444 when it spoke of "... those generic problems under continuing study which have.... potentially significant public safety implications." Six of the twenty-two tasks identified with the "Unresolved Safety Issues" are not applicable to Grand Gulf because they apply to pressurized water reactors only. These tasks are A-2, A-3, A-4, A-5, A-12, and A-26. Also, tasks A-6, A-7, and A-8 only apply to Mark I or Mark II boiling water reactor containments. With regard to the remaining 13 tasks that are applicable to Grand Gulf the NRC staff has issued NUREG reports providing its resolution of five of the issues. The table below lists those issues.

| Task Number | NUREG Report and Title | SER/SER Suppl. Section(s)* |
|---|---|---|
| A-10 | NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" | |
| A-24 | NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" | |
| A-31 | SRP 5.4.7 and BTP 5-1 "Residual Heat Removal Systems" incorporate requirements of USI A-31. | |
| A-36 | NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" | |
| A-42 | NUREG-0313, Revision 1 "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping" | |

*Not available at this time. To be provided by the Project Manager.

The remaining issues applicable to Grand Gulf are listed in the following table.

GENERIC TASKS ADDRESSING
"UNRESOLVED SAFETY ISSUES"
THAT ARE APPLICABLE TO
GRAND GULF UNITS 1 AND 2

| | | |
|---|---|---|
| 1. | A-1 | Water Hammer |
| 2. | A-9 | ATWS |
| 3. | A-11 | Reactor Vessel Materials Toughness |
| 4. | A-17 | Systems Interaction in Nuclear Power Plants |
| 5. | A-39 | Safety Relief Valve Pool Dynamic Loads |
| 6. | A-40 | Seismic Design Criteria |
| 7. | A-43 | Containment Emergency Sump Reliability |
| 8. | A-44 | Station Blackout |

With the exception of Tasks A-9, A-43, and A-44, Task Action Plans for the generic tasks above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." A technical resolution for Task A-9 has been proposed by the NRC staff in Volume 4 of NUREG-0460, issued for comment. This served as a basis for the staff's proposal for rulemaking on this issue. The Task Action Plan for Task A-43 was issued in January 1981, and the Task Action Plan for A-44 was issued in July 1980. The information provided in NUREG-0649 meets most of the informational requirements of ALAB-444. Each Task Action Plan provides a description of the problem; the staff's approaches to its resolution; a general discussion of the bases upon which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower required; a description of the interactions with other NRC offices, the Advisory Committee on Reactor Safeguards and outside organizations; estimates of funding required for contractor-supplied technical assistance; prospective dates for completing the tasks; and a description of potential problems that could alter the planned approach or schedule.

In addition to the Task Action Plans, the staff issues the "Aqua Book" (NUREG-0606) on a quarterly basis. This book entitled, "Office of Nuclear Reactor Regulation Unresolved Safety Issues Summary, Aqua Book," provides current schedule information for each of the "Unresolved Safety Issues." It also includes information relative to the implementation status of each "Unresolved Safety Issue" for which technical resolution is complete.

We have reviewed the eight "Unresolved Safety Issues" listed above and the four new USIs discussed in Section C.4 as they relate to Grand Gulf Units 1 and 2. Discussion of each of these issues including references to related discussions in the Safety Evaluation Report is provided below in Section C.5. We have satisfactorily concluded our review for all but the A-39, "Mark III Safety Relief Valve Pool Dynamic Loads" issue. That issue is currently incomplete. We will discuss resolution of this issue in a supplement to the Safety Evaluation Report. Based on our review of these items, we have concluded, for the reasons set forth in Section C-5, that with the exception of A-39 there is reasonable assurance that the Grand Gulf Unit Nos. 1 and 2 can be operated prior to the ultimate resolution of these generic issues without endangering the health and safety of the public.

## C.4 New "Unresolved Safety Issues"

An in-depth and systematic review of generic safety concerns identified since January 1979 has been performed by the staff to determine if any of these issues should be designated as new "Unresolved Safety Issues." The candidate issues originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident," ACRS recommendations, abnormal occurrence reports, and other operating experience. The staff's proposed list was reviewed and commented on by the ACRS, the Office of Analysis and Evaluation of Operational Data (AEOD) and the Office of Policy Evaluation. The ACRS and AEOD also proposed that several additional "Unresolved Safety Issues" be considered by the Commission. The Commission considered the above information and approved the following four new "Unresolved Safety Issues:"

A-45 Shutdown Decay Heat Removal Requirements

A-46 Seismic Qualification of Equipment in Operating Plants

A-47 Safety Implication of Control Systems

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

A description of the above process together with a list of the issues considered is presented in NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants, Special Report to Congress," dated March 1981. An expanded discussion of each of the new "Unresolved Safety Issues" is also contained in NUREG-0705.

The applicability and bases for licensing prior to ultimate resolution of the four new USIs for Grand Gulf Units 1 and 2 are discussed in Section C.5.

## C.5 Discussion of Tasks as They Relate to Grand Gulf

This section provides the NRC staff's evaluation of the Grand Gulf facilities for each of the applicable "Unresolved Safety Issues." This includes our bases for licensing prior to ultimate resolution of these issues. Our conclusions are based in part on information provided by the applicant in their letter of August 7, 1981 from L. F. Dale, Mississippi Power and Light Company to Robert L. Tedesco, NRC.

## 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 line.

Most of the damage reported has been relatively minor, 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.

Under generic Task A-1, the potential for waterhammer in various systems is being evaluated and appropriate requirements and systematic review procedures are being developed to ensure that waterhammer is given appropriate consideration in all areas of licensing review. A technical report, NUREG-0582, "Water-hammer in Nuclear Power Plants" (July 1979), providing the results of an NRC staff review of waterhammer events in nuclear power plants and stating staff licensing positions, completes a major subtask of Generic Task A-1.

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 reactor 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.

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. Further assurance for filled discharge piping is provided by pressure instrumentation at the piping high points. 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.

With regard to additional protection against potential waterhammer events currently provided on 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.

In addition, we require that the applicant conduct a preoperational vibration dynamic effects test program in accordance with Section III of the American Society of Mechanical Engineers Code for all Class 1 and Class 2 piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips, and other operating modes associated with the design operational transients.

Nonetheless, in the unlikely event that a large pipe break did result from a severe waterhammer 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 Task A-1 identifies potentially significant waterhammer scenarios which have not explicitly been accounted for in the design and operation of Grand Gulf, corrective measures will be required at that time. The task has not identified the need for measures beyond those already implemented.

Based on the foregoing, we conclude that Grand Gulf can be operated prior to ultimate resolution of the A-1 generic issue without undue risk to the health and safety of the public.

A-9 Anticipated Transient 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 (initiating 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 systems did not "scram" as desired, then an "anticipated transient without scram," or ATWS, would have occurred.

Grand Gulf has been required to provide a 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. Grand Gulf will have ATWS operator procedures and APT in place upon initial criticality.

The anticipated transient without scram issue is currently scheduled for rulemaking in summer 1981. The applicant will be required to comply with any further requirements on anticipated transient without scram which may be imposed as a result of the rulemaking.

Based on our review, we conclude that there is reasonable assurance that Grand Gulf can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

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.

The principal objective of Task A-11 is to develop safety criteria to allow a more precise assessment of safety margins during normal operation, transients and accident conditions in older reactor vessels with marginal fracture toughness.

Based on our evaluation of this facility's reactor vessels materials toughness, we have concluded that these units will have adequate safety margins against brittle failure during operating, testing, maintenance and anticipated transient conditions over the life of the units. Since Task A-11 is projected to be completed well in advance of this facility's reactor vessel reaching a fluence level which would notably reduce fracture resistance, acceptable vessel integrity for the postulated accident conditions will be assured at least until the reactor vessel is reevaluated for long-term acceptability, as will be required as our implementation requirement for Task A-11.

In addition, the surveillance program required by 10 CFR 50, Appendix H will afford an opportunity to reevaluate the fracture toughness periodically during the first half of design life.

Therefore, based upon the foregoing, we have concluded that Grand Gulf can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-17 Systems Interaction in Nuclear Power Plants

Currently licensing requirements are founded on the principle of defense-in-depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against hazards such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, human factors, and sabotage. These design provisions are subject to review against the Standard Review Plan (NUREG-75/087) which requires interdisciplinary reviews and addresses many different types of potential systems 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. Thus, the current licensing requirements and procedures provide for a degree of plant safety with respect to such interactions.

In November 1974, the Advisory Committee on Reactor Safeguards requested that the NRC staff give attention to the need to increase safety by separately evaluating the plant from a multidisciplinary point of view, in order to identify potentially undesirable interactions between plant systems. The concern arises because the design, analysis and installation of systems is frequently the responsibility of teams of engineers with functional specialties--such as civil, electrical, mechanical, or nuclear. Experience at operating plants led the ACRS to question whether the work of these functional specialists is sufficiently integrated to enable them to minimize adverse interactions among systems. Such adverse events have occurred because the teams did not assure by adequate coordination that the required independence of safety systems was provided under all conditions of operation.

In mid-1977, Task A-17 was initiated to assure that present review procedures and safety criteria provide an acceptable level of redundancy and independence for safety functions. The task proceeded by evaluating the potential for undesirable interactions between systems at a sample plant.

The NRC staff's current procedures assign primary responsibility for review of various technical areas to specific organizational units and assign secondary responsibility to other units where there is a functional interface. Designers follow somewhat similar procedures and provide the analyses of systems and interface reviews. Task A-17 provided an independent study of methods that could identify important systems interactions that adversely impact safety, and which were not considered by current review procedures. The first phase of this study began in May 1978 and was completed in February 1980 by Sandia Laboratories under contract to the NRC staff.

The Phase I investigation was structured to identify areas where interactions are possible between systems and have the potential of negating or seriously degrading the performance of safety functions. The study concentrated on commonly caused or linked failures among systems that could violate a safety function. The investigation was to then identify where NRC review procedures may not have properly accounted for these interactions.

The Sandia Laboratories used fault-tree methods to identify component failure combinations (cut-sets) that could result in loss of a safety function. The cut-sets were further reduced 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 the staff's 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 were not within the Phase I scope of the Sandia study.

The "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, provides for a systems interaction follow-on study, Section II.C.3, "Systems Interactions." Since April 1980, the Office of Nuclear Reactor Regulation has intensified the effort both by broadening the study of methods to identify potential systems interactions and by performing audit reviews of two plants for selected systems interactions. Our recent experience provides a basis from which we are developing an improved systematic review process for potential systems interactions. The process will provide for a resolution of USI A-17, assimilate operating reactor experience, and rank identified systems interactions by their relative importance to safety.

In addition to the staff's interdisciplinary review, the Grand Gulf 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.

In addition, the interface between Bechtel, General Electric, and Mississippi Power and Light is tracked by the Grand Gulf project control log.

To assure that all discipline interactions have identified all potential hazards to safety related equipment, the Grand Gulf 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 Grand Gulf Engineering Review Team:

- Non-Seismic Category I Over Seismic Category I
- High Energy Line Break
- Flooding
- Jet Impingement

Therefore, we conclude that there is reasonable assurance that Grand Gulf can be operated prior to the final resolution of this generic issue without endangering the health and safety of the public.

A-39 Safety/Relief Valve Hydrodynamic Loads

All BWR plants are equipped with a number of SRVs to control primary system pressure transients. The SRVs are mounted on the main steam lines inside the drywell with discharge lines routed through the drywell into the suppression pool. When an SRV is actuated the steam released from the primary system is discharged into the suppression pool where it is condensed.

Actuation of an SRV can be either automatic, at a preset pressure, or manual by means of an external signal. A preselected number of SRVs are used for the Automatic Depressurization System (ADS) which is designed to reduce the reactor pressure and permit operation of the low pressure emergency core coolant systems. The ADS performs this function by automatic actuation of the specified SRVs following receipt of specific signals from the reactor protection system.

Upon actuation of an SRV, the air column within the partially submerged discharge line is compressed by the high pressure steam and, in turn, accelerates the water leg into the suppression pool. The water jets thus formed create pressure and velocity transients which are manifested as drag or jet impingement loads on submerged structures.

Following water clearing, the compressed air is also accelerated into the suppression pool forming high pressure air bubbles. These bubbles execute a number of oscillatory expansions and contractions before rising to the suppression pool surface. The associated transients again create drag loads on submerged structures as well as pressure loads on the submerged boundaries. These loads are referred to as SRV air clearing loads. Containment structures, equipment and piping shall be designed to accommodate these loads.

In July 1976, the staff issued acceptance criteria for SRV loads for the Mark III containments. These criteria were established on the basis of our evaluation of the methodology for predicting the SRV loads which was proposed by the General Electric Company. In late 1980, however, GE proposed a revised method, which will result in substantial reduction of SRV loads. This improved method was based on the Caorso* inplant SRV tests which were performed in January 1979 in Italy. In addition, Grand Gulf has stated that they plan to perform in-plant confirmatory tests of their SRV quencher discharge. Grand Gulf has also used the revised SRV loads proposed by GE.

We are currently reviewing this new methodology for predicting the SRV loads. The results of our generic evaluation will be presented in a NUREG report which is currently scheduled to be issued in the fourth quarter of 1981. Our evaluation of the plant-specific application of this method for Grand Gulf will be reported in a Supplement to this SER.

A-40 Seismic Design Criteria - Short-Term Program

NRC regulations require 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. In addition, some revisions to sections of the Standard Review Plan and regulatory guides to bring them more in line with the state-of-the-art will result.

The seismic design basis and seismic design of Grand Gulf has been evaluated at the operating license stage using current licensing criteria and requirement. The staff's review of Grand Gulf to those criteria is discussed in Section ______ of this Safety Evaluation Report. Should the resolution of Task A-40 indicate a change is needed in these licensing requirements, all operating reactors including Grand Gulf will be re-evaluated on a case-by-case basis. Accordingly, we have concluded that Grand Gulf can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

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

*Caorso is a BWR/Mark II plant located in Caorso, Piancenza in Italy.

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.

One postulated means of losing the ability to draw water from the suppression pool could be blockage by debris. A principal source of such debris could be the thermal insulation on the reactor coolant system piping. In the event of a piping break, the subsequent violent release to the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. This debris could then be carried over into the suppression pool, potentially causing blockage.

A second postulated means of losing the ability to draw water from the suppression pool could be abnormal conditions at the pump inlet such as air entrainment or vortices. These conditions could result in pump cavitation, reduced flow and possible damage to the pumps. Due to the relatively low submergence for ECCS suction lines for Mark III containments (i.e., 4 ft. minimum submergence), the staff requires that the applicant perform in-plant preoperational tests at minimum suction submergence for each of the ECCS systems to demonstrate that circulation through the pool can be readily accomplished without significant vortex formation. We will condition the operating license for Grand Gulf that these tests be completed by the fuel load date.

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.

Accordingly, we conclude that Grand Gulf can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

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 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, even 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. Our review of the design, testing, surveillance, and maintenance provisions for the onsite emergency diesels is described in Section ______ of this SER. 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.

The issue of station blackout was considered by the Atomic Safety and Licensing Appeal Board (ALAB-603) for the St. Lucie Unit No. 2 facility. In addition, in view of the completion schedule for Task A-44 (October 1982), the Appeal Board recommended that the Commission take expeditious action to ensure that other plants and their operators are equipped to accommodate a station blackout event. The Commission has reviewed this recommendation and determined that some interim measures should be taken at all facilities including Grand Gulf while Task A-44 is being conducted. Consequently, interim emergency procedures and operator training for safe operation of the facility and restoration of alternating current power will be required. The staff notified the applicant of these requirements in a letter from D. Eisenhut, NRC, to the applicant dated ____________. We will condition the operating license for Grand Gulf that these procedures and this training be completed by fuel load date.

Based on the above, we have concluded that there is reasonable assurance that Grand Gulf can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

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 system 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.

Following the TMI accident, the industry performed and documented extensive analyses of feedwater transients and small-break loss-of-coolant accidents to support 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.

Based on the above, we have concluded that Grand Gulf can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

## 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 commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist 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.

Grand Gulf was reviewed against current seismic criteria and approved by the Commission staff in accordance with current design criteria and methods for seismic qualification. The staff's review is discussed in Section _________ of this Safety Evaluation Report. Therefore, we conclude that Grand Gulf can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

## 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 failurs 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 result 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 purpose of this "Unresolved Safety Issue" is to define generic criteria that will be used for plant-specific reviews.

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 non-safety systems or providing isolating devices between safety and non-safety systems. These devices preclude the propagation of non-safety 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 the 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 System 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 in 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.

Based on the above, we have concluded that there is reasonable assurance that Grand Gulf can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

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

Following a loss-of-coolant accident in 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 meterials by the spray solution; and (4) any synergistic chemical, thermal and radiolytic effects of post-accident enviornmental conditions on containment protective coating systems and electric cable insulation.

Becase of the potential for significant hydrogen generation as the result of an accident, 10 CFR 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," and Criterion 41 of the General Design Criteria. "Containment Atmosphere Cleanup," in Appendix A to 10 CFR Part 50, requires that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

The 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 smaller, low-pressure containments. As a result, the Commission determined that a rulemaking proceeding should be undertaken to define the manner and extent to which hydrogen evolution and other effects of a degraded core need to be taken into account in plant design. An advance notice of this rulemaking proceeding on degraded core issues was published in the Federal Register on October 2, 1980.

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 a second 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 applicant's evaluation of alternate hydrogen control measures was provided. A Hydrogen Ignition System (HIS) was selected and detailed evaluations of containment pressure and temperature response were performed.

The HIS 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 constitute 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.

The staff has reviewed and approved (1) the Grand Gulf Hydrogen Ignition System, and (2) the applicant's analysis of the ability of essential equipment to survive the hydrogen burn environment. This evaluation is provided in Sections ________ and ________ of this Safety Evaluation Report.

Based on the above, we conclude that Grand Gulf can be operated prior to resolution of the "Unresolved Safety Issue" and the proposed rulemaking without undue risk to the health and safety of the public.