UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'83 APR 18 A10:08

Before the Atomic Safety and Licensing Board

OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH

In the Matter of	)	
	)	
Gulf States Utilities Company, *et al*.	)	Docket Nos. 50-458 50-459
	)	
(River Bend Station, Units 1 and 2)	)	

APPLICANTS' ANSWER TO THE CONTENTIONS FILED BY PETITIONERS FOR LEAVE TO INTERVENE, STATE OF LOUISIANA THROUGH THE OFFICE OF THE ATTORNEY GENERAL, LOUISIANA CONSUMER'S LEAGUE, LOUISIANANS FOR SAFE ENERGY, INC. AND GRETCHEN REINEKE ROTHCHILD

Introduction

By a Memorandum and Order dated July 30, 1982, the presiding Atomic Safety and Licensing Board ("Licensing Board" or "Board") held that petitioners for leave to intervene, namely, the State of Louisiana,[1/] Louisiana Consumers League, Inc., Louisianans for Safe Energy, Inc. and Gretchen Reineke Rothchild, had satisfied the interest portion of the Nuclear Regulatory Commission's regulations relating to intervention. Accordingly, the Licensing Board

---

1/ The State of Louisiana through the Office of the Attorney General had petitioned to participate under the provisions of both 10 C.F.R. §§2.714 and 2.715. By Memorandum and Order Ruling on Petitions to Intervene (February 10, 1982), the Board had admitted the State under the provisions of 10 C.F.R. §2.715(c).

DS03

directed petitioners to submit proposed contentions by September 7, 1982. By an August 20, 1982 Order, the Board extended the time for filing contentions to December 13, 1982. In a subsequent Order of December 21, 1982, the Board again extended the time for filing contentions to March 15, 1983.

On December 15, 1982, a pleading entitled "Contentions By Joint Intervenors [Louisiana] Consumers' League, Inc.[,] Louisianans for Safe Energy[,] Gretchen Reineke Rothchild" was submitted.[2/] On March 15, 1983, the "Supplemental Petition of the State of Louisiana" was filed, superseding an earlier submittal of contentions dated December 15, 1982.

As discussed below, Applicants, Gulf States Utilities, et al. ("GSU" or "Applicants"), submit that all of the contentions filed are either inadequate under the Commission's rules or are premature. Thus, the Licensing Board should deny the pending petitions for leave to intervene.

I. General Discussion of the Rules Governing Admissibility of Contentions

It is well established that a contention which merely alleges generally that the application for an operating

---

2/ Petitioners have inaccurately referred to themselves as "Joint Intervenors." Applicants note that the Board has not determined that the requirements of 10 C.F.R. §2.714 for filing contentions have been satisfied, and petitioners have therefore not achieved status as intervenors. Accordingly, Applicants will refer to them as "petitioners."

license is inadequate fails to meet the requirement of specificity contained in 10 C.F.R. §2.714(b). Licensing boards have refused to admit such contentions. This is particularly true where the Final Safety Analysis Report, Environmental Report-Operating License Stage, or other portion of the application discusses in detail the manner in which the Commission's requirements are being met.

For example, in Susquehanna [3/] the Licensing Board denied a contention which alleged that portions of the applicant's environmental report understated certain effects of an accident. The Board held that the proposed contention failed to meet the Commission's test for specificity and stated:

> In order to evaluate whether a contention presents an issue in controversy, the regulations specify that their bases should be set forth with reasonable specificity. Here we are left to wander aimlessly in our speculation on the details of the allegations--a practice obviously unfair to proper procedure, to the parties and the Board. The contention will not be admitted. 4/

Likewise, in Offshore Power [5/] the Licensing Board

3/ Pennsylvania Power & Light Company (Susquehanna Steam Electric Station, Units 1 and 2), Docket Nos. 50-387 and 50-388, "Memorandum and Order on Pending Motions and Requests" (July 7, 1981).

4/ Id., slip op. at 4.

5/ Offshore Power Systems (Manufacturing License for Floating Nuclear Power Plants), LBP-77-48, 6 NRC 249 (1977).

denied admission of a contention which merely asserted that the Staff in the Final Environmental Statement inadequately considered and improperly dismissed various alternatives to the licensing of the proposed stations. The Board found this contention to be inadmissible because it was conclusionary and lacking in the necessary specificity and factual bases.[6/] Further, the Board stated that "[c]ontentions which are barren and unfocused are of no assistance to us in the resolution of the issues to be decided." 7/ In the Seabrook proceeding, the Licensing Board rejected a contention regarding a general deficiency in the qualification of equipment as too broad to be litigated in an operating license proceeding because the intervenor had not specifically designated the equipment or categories of equipment to which the contention related.[8/]

Similarly, in the G.E. Morris [9/] proceeding, the Licensing Board rejected a contention that the applicant had not taken into account the close proximity of two

---

6/ Id. at 250.

7/ Id. at 251.

8/ Public Service Company of New Hampshire (Seabrook Station, Units 1 and 2), Docket Nos. 50-443 OL and 50-444 OL, "Memorandum and Order" (September 13, 1982) (slip op. at 15-16).

9/ General Electric Company (GE Morris Operation Spent Fuel Storage Facility), Docket No. 70-1308 OLA (Spent Fuel Pool), "Order Ruling on Contentions of the Party" (June 4, 1980).

facilities, noting that "[n]o litigable issue is presented" by the mere recitation of this information in the petition.[10/] See also Commonwealth Edison Company (Dresden Nuclear Power Station, Unit No. 1), Docket No. 50-10-OLA, "Memorandum and Order" (July 12, 1982) (slip op. at 9); Public Service Company of Oklahoma (Black Fox Station, Units 1 and 2), Docket Nos. STN 50-556CP and 50-557CP, "Memorandum and Order" (January 11, 1982) (slip op. at 2-4); Long Island Lighting Company (Shoreham Nuclear Power Station, Unit 1), LBP-81-18, 14 NRC 71, 75 (1981).

Thus, those contentions which merely allege inadequacies or violations in the broadest of terms do not meet the standards of Section 2.714(b) for specificity and bases. The proposed contentions in this proceeding should therefore be denied.

Further, several of petitioners' proposed contentions are premature and must be denied or deferred pursuant to the Appeal Board's recent ruling in Catawba.[11/] In that proceeding, the Appeal Board considered the circumstances under which the Licensing Board may allow the conditional admission of a contention subject to later specification.

10/ Id., slip op. at 20.

11/ Duke Power Company (Catawba Nuclear Station, Units 1 and 2), ALAB-687, 16 NRC __ (August 19, 1982).

The Board held that such "conditional" contentions were not to be accepted and stated:

> Nothing in the terms of Section 2.714(b) explicitly vests a licensing board with the power to admit an unacceptably vague or imprecise contention conditionally, subject to later revision upon receipt of additional information. Rather, as we read it, the Section conveys the clear message that, in order to be admitted, the contention must meet the "requirements of this [Section]"; i.e., it must set forth its bases "with reasonable specificity." Moreover, the administrative history of the Section precludes any suggestion that the Commission intended an implicit exception to the specificity requirements in circumstances where, because of a lack of available information, it is not possible for the petitioner to meet those requirements at the time its contentions are due. 12/

In applying this ruling to the contentions before it, the Licensing Board in Catawba held that contentions relating to emergency planning were premature because the plans were not available.[13/] The Licensing Board rejected those contentions which lacked the requisite specificity, but acknowledged that it would also be within the Board's discretion to defer such contentions. Under the rulings in Catawba, however, it is not within the discretion of the

---

12/ Id., slip op. at 9.

13/ Duke Power Company, supra, Docket Nos. 50-413 and 50-414, "Memorandum and Order (Reflecting Decisions Made Following Second Prehearing Conference)" (December 1, 1982) (slip op. at 5).

Board to admit such premature contentions. Petitioners' proposed contentions which are subject to this defect should therefore be denied.

Utilizing these general principles, each of the proffered contentions is discussed below. Because a number of contentions of the joint petitioners and the State overlap, they are discussed together. Where there is a general allegation, the subject matter of which is discussed in detail in the application, this fact is noted, not to refute the merits of the contention, but to indicate the lack of specificity and the absence of any litigable issue.

II. Specific Discussion of Proffered Contentions

Joint Petitioners' Contention 1
Financial and Technical Qualifications

Joint petitioners attempt to make a showing of "special circumstances" pursuant to 10 C.F.R. §2.758(b) in order to overcome the Commission regulation which prohibits consideration of an applicant's financial qualifications in licensing proceedings.[14] Petitioners have failed to cite any such special circumstances which would justify a waiver of this regulation in this proceeding and, in fact, the arguments they offer were rejected by the Commission in its rulemaking proceeding for the adopted regulation. Nor have

---

14/ 10 C.F.R. §50.40(b).

petitioners complied with the mandatory procedures for seeking a waiver of regulation.

As noted, the sole ground for a challenge to any Commission rule or regulation in an adjudicatory proceeding is that "special circumstances" with respect to the subject matter of the particular proceeding are such that application of the rule or regulation would not serve the purpose for which it was adopted. Such a request must be accompanied by an affidavit that identifies specific aspects of the subject matter of the proceeding as to which application of the rule or regulation would not serve the purposes for which it was adopted. It must set forth with particularity the special circumstances alleged to justify the waiver or exception requested. Unless a petitioner makes a prima facie showing that the challenged rule or regulation would not serve the purpose for which it was adopted, no evidence on the issue can be received and the presiding officer may not consider the matter further.[15/]

No affidavit as required has been submitted by petitioners. As "special circumstances," they merely allege that GSU's financial status has changed substantially for the worse since the construction permit was granted,[16/]

15/ 10 C.F.R. §2.758(c), (d).

16/ Contentions By Joint Intervenors Consumers' League, Inc.[,] Louisianans for Safe Energy[,] Gretchen Reineke Rothchild ("Contentions by Joint Intervenors") at 2

that there have been "numerous inspection reports documenting where construction activities were not conducted in full compliance with regulatory requirements under circumstances indicating that cost-cutting measures were involved."17/ The Commission has addressed and rejected virtually the same arguments in the rulemaking proceeding;18/ thus they cannot constitute the showing required by 10 C.F.R. §2.758.

Underlying their proposed contention is petitioners' erroneous assumption that the general ability of a utility to finance construction of new generating facilities is inevitably linked to considerations of the public health and safety under the Atomic Energy Act of 1954, as amended, 42 U.S.C. §2011 et seq. and implementing regulations. The Commission expressly rejected this view in eliminating the financial qualifications rule "because of the lack of any demonstrable link between public health and safety concerns and a utility's ability to make the requisite financial showing."19/

(December 13, 1982).

17/ Id.

18/ See generally 47 Fed. Reg. 13750 (March 31, 1982) (codified in 10 C.F.R. §50.40(b)).

19/ Id. at 13751.

In a recent decision, the Licensing Board in Byron [20/] considered and rejected the admission of precisely the issues sought to be raised here. The Board disposed of the allegation that the applicant did not possess or have reasonable assurance of obtaining funds necessary to complete construction, operation and decommissioning of the facility safely by noting that since "[n]othing was provided to show that a lacking of financial qualifications *ipso facto* translated itself into a threat to public health and safety," a general allegation of insufficient funds fails to establish the requisite special circumstances.[21/]

In order to qualify for a waiver of the regulation, the Licensing Board in Byron held that a petitioner must affirmatively demonstrate:

> . . . an actual causal connection between a utility's financial qualifications and it presenting a threat to public health and safety . . . . Without the establishment of a demonstrable link between financial qualifications and it constituting a threat to public health and safety, there would be no showing that the subject regulations do not serve the purpose for which they were adopted. 22/

20/ Commonwealth Edison Company (Byron Station, Units 1 and 2), Docket Nos. STN 50-454 OL and STN 50-455 OL, "Memorandum and Order" (August 2, 1982).

21/ *Id.*, slip op. at 4.

22/ *Id.*, slip op. at 5.

Similarly, in this proceeding, petitioners' bare assertions that GSU's financial status has changed fail to provide the requisite showing and therefore should be rejected.

Joint petitioners allege that "numerous inspection reports" support their position, but cite only three.23/ Here, joint petitioners have failed to make any connection whatsoever between the three cited inspection reports and GSU's financial condition. One report relates to storage procedures, the second to the reportability of certain matters and the third to grades of reinforcing steel. In the Byron proceeding, supra, the Board expressly addressed the issue of similar, but more numerous noncompliances and concluded that there was no showing that the deficiencies were in any way related to applicant's financial condition.24/ Similarly, the Commission in the rulemaking proceeding noted that violations detected by the NRC Office of Inspection & Enforcement cannot, as a general matter, be "shown to arise from a licensee's alleged lack of financial qualifications."25/ No showing has been made that these are other than minor, isolated noncompliances which are not unexpected on a project of this magnitude.

23/ Contentions by Joint Intervenors at 2.

24/ Byron, supra, slip op. at 7.

25/ 47 Fed. Reg. at 13751.

The final item cited by petitioners relates to a request regarding one-half inch tubing.[26/] The request to allow alternate methods to meet Commission requirements was made to the NRC publicly and openly. NRC approval was therefore sought in the normal course of review. The NRC memorandum to which the joint petitioners refer reflects a meeting it held with GSU on October 22, 1982. The memorandum states, inter alia, that "the NRC Staff expressed a willingness to assist GSU in realizing its cost-effectiveness goals by responding in a reasonably short time-frame to such proposals from GSU."[27/] Inasmuch as GSU fully supported its proposed action and provided the NRC with a clear and supportable basis for accepting its proposal,[28/] there is no basis or support for petitioners' allegation that any adverse effect upon the public health and safety could result.

Having "failed to establish a nexus"[29/] between Applicants' financial condition and any of the four items

---

26/ Contentions by Joint Intervenors at 2.

27/ Memorandum for A. Schwencer from J. Stefano at 2 (November 3, 1982). Rather than addressing the costs associated with the change, the quoted passage relates to obtaining an early determination as to the acceptability of the change so the Applicants can know which course to pursue.

28/ Id.

29/ Byron, supra, slip op. at 7.

discussed above, joint petitioners have failed to make the requisite showing that the rule on financial qualifications should be waived. This contention should therefore be denied.

Joint Petitioners' Contention 2
Environmental Qualification

This contention alleges that "[t]he Applicants have not demonstrated that they will be in compliance"30/ with interim NRC Staff positions related to environmental qualification of certain equipment for the River Bend Station. This proposed contention raises no litigable issue inasmuch as it merely engages in prohibited speculation regarding Applicants' compliance with NRC requirements.

Section 3.11 of the Final Safety Analysis Report addresses with specificity the details of the Applicants' program to assure compliance with environmental qualifications requirements. In particular, page 3.11-3 addresses compliance with NUREG-0588, the document to which joint petitioners refer.31/ Section 3.11 also refers to an additional document entitled "Environmental Qualification Document,"32/ which is a constituent part of the River Bend Station Final Safety Analysis Report. This document

---

30/ Contentions by Joint Intervenors at 3.

31/ Id.

32/ Final Safety Analysis Report at 3.11-1.

addresses specific plant equipment and the program for its environmental qualification.

Moreover, on January 21, 1983, the Nuclear Regulatory Commission published in the Federal Register a final rule, entitled "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."[33/] This amendment codifies in 10 C.F.R. §50.40 the environmental qualification methods and criteria that meet the Commission's requirements in this area for nuclear power plants. Joint petitioners have pointed to no deficiency in the Final Safety Analysis Report, including the Environmental Qualification Document, nor any inability on the part of the River Bend Station to conform to the requirements of the Commission's rules regarding environmental qualification. Thus, this speculative contention is barren and unfocused and should be denied.

Joint Petitioners' Contention 3
Induced Seismic Activity

Joint petitioners allege that Applicants have failed to consider adequately the effects of seismic activity on the River Bend Station resulting from exploratory and/or production natural gas wells within the radius of the exclusion zone and from subsidence due to withdrawal of water, oil

---

33/ 48 Fed. Reg. 2729 (January 21, 1983) (codified at 10 C.F.R. Part 50).

and/or gas.[34/] This proposed contention is merely a generalized assertion, without any basis whatsoever, that Applicants have failed to consider induced seismic activity. This contention ignores completely the information and analyses contained in the Final Safety Analysis Report regarding the lack of induced seismic activity. In particular, Sections 2.2.3.1.2, 2.5.1.1.6 (particularly at page 2.5-38), 2.5.4.1.1, 2.5.1.2.8.6, and 2.5.1.2.3.4 treat these matters in detail.

Significantly, joint petitioners' proposed contention contains a serious misstatement of fact. They imply that there are exploratory or production natural gas wells "within the pertinent radius of the exclusion zone."[35/] To the contrary, as discussed in FSAR Section 2.1.2.1, GSU has ownership of mineral rights within the exclusion area, and there is no such activity presently being conducted (nor contemplated) within the designated exclusion area for the River Bend Station. Any statement or implication to the contrary in the proposed contention is entirely lacking in basis. If, at any time in the future such activity were to be considered, Applicants would appropriately notify the NRC. However, to assert that such activity might take place in the future is speculative and need not be considered by

---

34/ Contentions by Joint Intervenors at 3.

35/ Id.

this Licensing Board.[36/] Thus, no specific deficiency in the application has been demonstrated and this contention should therefore be denied.

Joint Petitioners' Contention 4
Prematurity of Application

Joint petitioners allege that Applicants have failed to provide the information required by 10 C.F.R. Parts 50 and 51 because they filed their application for an operating license "too early into its construction and planning process."[37/] No litigable issue is raised by this contention. The contention is based upon an apparent misunderstanding of the regulatory process and the interaction of the NRC Staff and Licensing Board.

The question of the acceptability for docketing of an application for an operating license, including the Final Safety Analysis Report and Environmental Report, is not a matter within the purview of this Licensing Board. Rather, Commission regulations assign this matter to the NRC Staff, which in this instance has accepted the application for docketing. Section 2.101(a)(2) of the NRC's regulations states that in order to determine whether such application is "complete and acceptable for docketing," an application

---

36/ Should there at any time in the future be exploratory or production natural gas wells within such zone, the Commission's regulations, notably 10 C.F.R. 52.206, provide adequate means of relief.

37/ Contentions by Joint Intervenors at 3.

for an operating license for a production or utilization facility will be initially treated as a tendered application after it is received. Section 2.101(a)(3) states that this determination is to be made by the Director of Nuclear Reactor Regulation, which necessarily means that this Licensing Board does not have any authority over the docketing of the application.38/

In any event, petitioners have made no showing that the NRC Staff acted improperly in docketing the application. It is important to note that NRC procedures contemplate that not all information must be furnished at the time an application for an operating license is docketed. The NRC recognizes that, during the course of the review, additional information will be submitted to complete the application and to respond to Staff inquiries.39/ The Commission has

---

38/ See Offshore Power Systems (Floating Nuclear Power Plants), ALAB-489, 8 NRC 194, 202 n.25 (1978).

39/ The technical specifications are noted as an example of information which has not been submitted. However, the final technical specifications which are issued by the NRC result from the review of the remainder of the application by NRC Staff and only come into focus near the end of this review process. Further, the NRC has promulgated a set of standard technical specifications for the guidance of applicants. This further diminishes the need for submission of technical specifications by an applicant at the outset of the review process.

recognized that this practice is not inconsistent with the rights of intervenors.[40/]

Thus, this contention raises no factual matters which may be litigated as a contention and it should be denied.

State of Louisiana's Contention 3
Joint Petitioners' Contention 5
Release of Radioactive Material Through Liquid Pathways

It is alleged that Applicants have failed to consider the effect of a release of radioactive material into surface and ground drinking water supplies.[41/] The sole basis of these contentions is a general reference to the Mississippi River and an aquifer in the Baton Rouge area. Neither the bases nor the contentions states with specificity any inadequacy in the Final Safety Analysis Report with regard to the discharges from the River Bend Station. In fact, Sections 2.4.12 and 2.4.13 of the FSAR describe in detail the accident analyses for the River Bend Station related to possible contamination of drinking water sources. Furthermore, FSAR Section 15.7.3 contains an assessment of

---

40/ See generally 10 C.F.R. §2.714(a)(1) (filing of late contentions); Catawba, supra, ALAB-687, slip op. at 14-18; Catawba, supra, "Memorandum and Order" (December 1, 1982) (slip op. at 2-7). In an Order dated December 23, 1982, the Commission indicated that it would review this aspect of the ruling in ALAB-687.

41/ Contentions by Joint Intervenors at 3-4; Supplemental Petition of the State of Louisiana at 5-6 ("Supplemental Petition").

accidental liquid releases into the Mississippi River as affecting surface water uses discussed in Environmental Report-Operating License Stage ("EROLS") Section 2.3.2.2 and groundwater uses discussed in Section 2.3.2.1. Furthermore, Section 5.4 of the EROLS discusses planned releases, including compliance with Appendix I to 10 C.F.R. Part 50.

In the Catawba case, the Licensing Board rejected a similar contention that "drinking water of communities downstream . . . will become contaminated by radioactive materials accidentally released from Catawba."[42/] The Licensing Board noted that the proposed contention did not even reflect awareness of the discussion of the facility's liquid radwaste system in the FSAR, including its analyses of possible accidents and their effects. The Board therefore concluded that the "vagueness of this contention provides no basis for arguments about the source or nature of the radioactive materials"[43/] or how downstream drinking water would be affected, and the contention was accordingly denied.

In the Seabrook proceeding, the Licensing Board similarly excluded a liquid pathway contention as failing to state why special treatment of liquid pathways should be

---

42/ Duke Power Company (Catawba Nuclear Station, Units 1 and 2), LBP-82-16, 15 NRC 566, 588 (1982).

43/ Id.

required.[44/] Similarly, petitioners here have failed to identify any inadequacy in the treatment of accidental releases in the FSAR or EROLS for River Bend and have therefore failed to specify any particular issue for litigation. Furthermore, petitioners have postulated no mechanism for discharges into the Baton Rouge regional aquifer.[45/] Thus, this matter lacks specificity and bases and should be denied. To the extent petitioners seek to include matters beyond the design bases accidents as discussed in the River Bend FSAR, the proposed contention is prohibited by the Commission's recent Policy Statement on Safety Goals.[46/] This contention should be denied.

Joint Petitioners' Contention 6
Generic Safety Issues

Joint petitioners assert that Applicants have failed to provide an adequate plan with respect to three unresolved safety issues which the NRC identified as a result of

---

44/ Public Service Company of New Hampshire (Seabrook Station, Units 1 and 2), 50-443-OL and 50-414-OL, "Memorandum and Order" (September 13, 1982) (slip op. at 13-14).

45/ To the contrary, EROLS Section 2.3.1.2 discusses that no such mechanism exists.

46/ See Safety Goal Development Program, 48 Fed. Reg. 10772 (March 14, 1983). See also Pennsylvania Power & Light Company (Susquehanna Steam Electric Station, Units 1 and 2), LBP-79-6, 9 NRC 291, 323 (1979) (denying contention postulating that "Class 8" accidents involving pipe breaks are more likely to occur than as indicated in applicants' Environmental Report).

investigations of the Three Mile Island accident.[47/] Petitioners err in stating that these matters are not addressed on the River Bend docket. In a letter from the NRC dated November 19, 1981, a copy of which is attached for the convenience of the Licensing Board, the NRC generic Issues Branch requested additional information on the status of unresolved safety issues pertaining to the River Bend Station. By letter dated June 3, 1982, Applicants submitted a response indicating in some detail the basis for proceeding with the licensing of the River Bend Station should these generic issues remain unresolved at the time an operating license is to be issued for the Station. Included within such discussion were the three items, Tasks A-45, A-47 and A-48, raised by joint petitioners. Moreover, as encouraged by the NRC, GSU has indicated its intent to adopt the positions of Licensing Review Group II[48/] as regards the resolution of these three generic safety issues.

As the Board is well aware, the unresolved generic safety issues will be addressed by the Staff in its Safety Evaluation Report. Regardless of any contention, the Licensing Board is obliged to review the Staff's approach to every identified unresolved generic safety issue to

---

47/ Contentions by Joint Intervenors at 4.

48/ Licensing Review Group II is an *ad hoc* group which includes utilities that own plants similar in design and licensing status to the River Bend Station.

determine its plausibility and sufficiency.[49/] Petitioners have made no allegation that the Applicants' commitment with regard to any unresolved generic safety issue will be unsatisfactory, and any such allegation prior to the availability of the Staff's Safety Evaluation Report would be premature at best.[50/] There is no basis for the assertion that these three matters are not addressed in the application, nor is there any indication as to why these positions are in any way inadequate. Thus, this contention should be denied.

Joint Petitioners' Contention 7
Cracking of Materials

Joint petitioners allege that Applicants have not demonstrated that River Bend Units 1 and 2 meet the requirements of 10 C.F.R. Part 50 Appendix A, General Design Criteria 4, 14, 30 and 31 "with regard to the adequacy of material selection and control and systems design."[51/] Specifically, in subsection (A) petitioners allege that the use of appropriate materials and processes as specified by NUREG-0313, Rev. 1[52/] has not been fully followed. This

---

49/ See, e.g., Jersey Central Power and Light Company (Oyster Creek Nuclear Generating Station), ALAB-645, 13 NRC 1024 (1981).

50/ Seabrook, supra, slip op. at 26-27.

51/ Contentions by Joint Intervenors at 4.

52/ Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping -

contention is lacking in bases. In response to Generic Letter 81-03, GSU performed an engineering assessment of the ASME Code Class 1, 2 and 3 stainless steel piping as outlined in NUREG-0313, Rev. 1. The results of this review are contained in letters dated September 4, 1981 and December 18, 1981. These two documents demonstrate compliance with the referenced generic letter and the recommendation of NUREG-0313 (Rev. 1). Thus, Part A of the proffered contention is without basis.

In Subsection B, it is asserted that the recommendations contained in NUREG-0619,[53/] relating to the installation of a low-flow controller to be used to control the feedwater flow over a range of flows has not been adequately implemented. Contrary to the assertion, the River Bend design incorporates a low flow controller to be used to control the feedwater flow over a range of flows.[54/] The response to Question 410.18[55/] addresses compliance with

---

Resolution to Generic Technical Activity A-42 (October 1979).

53/ BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking - Resolution to Generic Technical Activity A-10 (November 1980).

54/ *See* FSAR Figure 10.4-7 at coordinates F-2.(valve F002). Subsequent revision of the text of the FSAR will describe the function of this valve in more detail.

55/ FSAR at Q&R 4.6-3.

NUREG-0619 as to control rod drive nozzles.[56/] Thus, the River Bend Station is in agreement with NUREG-0619. Subsection (B) of this contention is also without basis and should be denied.

State of Louisiana's Contention 4
Joint Petitioners' Contention 8
Old River Control Structure

In these two similar contentions, it is alleged that Applicants have not adequately considered the effect of the failure of the Old River Control Structure on the safe operation of the River Bend Station.[57/] It is alleged that an "antiquated and structurally questionable barrier known as the Old River control structure"[58/] could fail thus allowing diversion of Mississippi River water to the Atchafalaya River. The asserted consequence would cause a significant decrease in the amount of water flowing through the natural course of the Mississippi River. It is alleged

---

56/ See also Section 4.6.1.1.2.4.2.4 of the FSAR. As the Appeal Board has stated, "an intervention petitioner has an ironclad obligation to examine the publicly available documentary material pertaining to the facility in question with sufficient care to enable it to uncover any information that could serve as the foundation for a specific contention." Catawba, supra, ALAB-687, slip op. at 13.

57/ Contentions by Joint Intervenors at 5; Supplemental Petition at 6. The joint petitioners' contention is even less specific than the State's.

58/ Supplemental Petition at 7.

that such a failure could have consequences on the safe operation of River Bend Station, Unit 1.

Initially, it must be emphatically noted that as thoroughly discussed in the FSAR Sections 9.2.5 and 9.2.7, the design of the River Bend Station is such that it places no reliance upon Mississippi River water as the ultimate heat sink for the Station. The Station utilizes a Category I cooling tower which stores sufficient water in its basin to shut down the facility and maintain shutdown for an extended period of time. This system utilizes deep wells and not Mississippi River water for makeup. Thus, the Old River Control Structure could not possibly have any effect on the safe shutdown capability of the plant.

In any event, it is entirely speculative that the control structure would be permitted to deteriorate to such a state that it would collapse. Indeed, the Federal courts have rejected the argument that such conjecture is necessary in complying with the National Environmental Policy Act of 1969, 42 U.S.C. §4321, _et seq_. _See_, _e.g._, _Warm Springs Dam Task Force v. Gribble_, 621 F.2d 1017, 1026 (9th Cir. 1980) (evaluation of possible dam failure not required). The structure has such a significant effect upon all downstream users, including the cities of Baton Rouge and New Orleans, that the effects on the River Bend Station of its failure would be _de minimus_ by comparison.

There is absolutely no reason given why the NRC cannot license this facility under the present flow regime in the

Mississippi River, or must assume hypothetically the collapse of the Old River Control Structure. Because safe operation of the River Bend Station would not be affected by the hypothetical loss of the Old River Control Structure, there is really no issue which this Licensing Board can consider nor any appropriate relief which it could grant.

The loss of the Old River Control Structure as a contingency on the normal operation of the River Bend Station could only be considered based upon the actual circumstances if that hypothetical event were to occur. Any potential impacts which operation of the River Bend Station might have upon river flows would have to be considered together with all other impacts by the cognizable officials at that time. To single out River Bend at this time for hypothetical treatment of this speculative event would serve no useful purpose.

As to consideration of the particular impacts alleged, it is noted that matters related to thermal discharges are beyond the jurisdiction of the NRC[59/] and would be resolved by the appropriate NPDES permit issuing authority for discharges occurring within the State of Louisiana which is at this time the United States Environmental Protection Agency. Finally, the two contentions are entirely lacking

---

59/ Public Service Company of New Hampshire (Seabrook Station, Units 1 and 2), CLI-78-1, 7 NRC 1, 26 (1978).

in specificity with regard to the allegation that the salt content of Mississippi River water would be substantially increased should the Old River Control Structure fail.

Thus, Applicants submit that these contentions raise no litigable matter before the Licensing Board and that such contentions should be denied.

State of Louisiana's Contention 2
Joint Petitioners' Contention 9
Emergency Response Plan

In these two contentions,[60/] the State of Louisiana and the joint petitioners allege certain deficiencies in the emergency planning efforts that the State, parishes and Applicants are making with regard to the River Bend Station. By way of background, it should be noted that the State of Louisiana has already published and placed in operation the State of Louisiana Peacetime Radiological Response Plan. This plan has undergone review by both the Federal Emergency Management Agency ("FEMA") and the NRC in the context of two completed operating license reviews. It is this plan which is referenced in the present Final Safety Analysis Report for the River Bend Station. Furthermore, the State of Louisiana has extensive experience in the implementation of emergency planning, including large-scale evacuation, as a

60/ Contentions by Joint Intervenors at 5-7; Supplemental Petition at 2-5.

result of a number of non-nuclear occurrences within the State.

Applicants have been working closely with the State of Louisiana and the involved parishes in order to complete the site and the plume exposure pathway Emergency Planning Zone ("plume EPZ") plans for the River Bend Station.[61/] It is expected that such planning efforts will be completed by the end of 1983, culminating in the submission of the completed plans for the River Bend Station. These plans include a revised State of Louisiana Peacetime Radiological Response Plan, which, in turn, will include a specific attachment for planning for the River Bend Station and the involved parishes. Additionally, designation of the plume EPZ by the officials having jurisdiction is expected to be complete and documented by July, 1983.

Thus, considerable effort has been made in the area of emergency planning for the River Bend Station, and no obstacle to the completion of an emergency plan has been encountered. However, in accordance with the Appeal Board's decision in ALAB-687, Applicants submit that it is premature to entertain general contentions related to emergency planning.

61/ In addition, planning for the ingestion pathway EPZ is also underway.

Nonetheless, there are certain parts of the contentions related to emergency planning submitted by the State of Louisiana and the joint petitioners which are impermissible as contentions under the Commission's rules. Applicants submit that these items should be denied now by the Licensing Board so that permissible contentions, if any, in specific areas of emergency planning can be quickly brought into focus when the emergency planning submittals are made.

For example, joint petitioners allege in subsection (A) that the proposed plant site is in close proximity to the Mississippi River and a regional aquifer, and that a reactor meltdown at River Bend will endanger these drinking water sources. This contention has no apparent relevance in the context of emergency planning and seems to be a reassertion of joint petitioners' proposed Contention 5. As discussed above, the Commission's Policy Statement on Safety Goals makes it clear that accidents beyond design basis are not to be considered in the licensing process. This includes planning for emergencies under 10 C.F.R. §50.47 and Appendix E to 10 C.F.R. Part 50.[62/] Alternatively, petitioners may be raising a siting issue which, as discussed previously, is inappropriate at the operating license stage. Therefore, admission of this proposed contention would be improper under the Commission's rules.

62/ 48 Fed. Reg. at 10779.

As a basis for this contention, joint petitioners allege that "[t]he Capital Area Groundwater Conservation Commission has petitioned the U.S. Environmental Protection Agency to declare the Baton Rouge Aquifer a sole source aquifer under the Federal Safe Drinking Water Act."[63/] The legal authority for this petition apparently is Section 1424(e) of the Safe Drinking Water Act.[64/] A reading of this section reveals that it is related to the interim regulation of underground injections into a sole source aquifer. For River Bend, no such injections are contemplated.[65/] Thus, this proposed contention is defective and should be denied.

Subsection (B) of joint petitioners' proposed contention alleges that the number, location, and capacity of local sheltering facilities and the degree of protection from radionuclides afforded thereby are inadequate.[66/] Joint petitioners apparently do not understand the Commission's requirements with regard to "sheltering facilities." Relocation centers must be provided outside the plume exposure EPZ to house evacuees from evacuated areas.

63/ Contentions by Joint Intervenors at 6.

64/ 42 U.S.C. §300h-3(e).

65/ 42 U.S.C. §300h(d)(1) defines the term "underground injection" as meaning "the subsurface emplacement of fluids by well injection."

66/ Contentions by Joint Intervenors at 6.

Such facilities are to be located at least five miles from the perimeter of the plume exposure EPZ.[67/] Thus, it is clear under the Commission's regulations that relocation facilities would not be required to afford "sheltering" in the sense of protection from radionuclides released from the Station in the event of an accident.

Rather, the protective action of "sheltering" during an emergency would entail instructions by authorities to residents within the plume exposure EPZ or designated portions to stay in their homes. This protective action would be chosen over other alternatives if appropriate to minimize doses to the individual, taking into account actual emergency conditions including protective sheltering factors as they exist. Neither the Commission's regulations nor guidance require the construction of special "fallout" type shelters within the plume exposure EPZ. Thus, this contention which alleges that additional protection is required lacks any basis and should be denied.

In subsection (C), joint petitioners attempt to raise the matter of "[t]he heightened sensitivity to radiation of children and pregnant women over that of the the average

---

67/ NUREG-0654 (Rev. 1), Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants at 63 (November 1980).

healthy adult male."[68/] This proposed contention is an attack upon the Commission's regulations which adopt the EPA Protective Action Guides for initiation of protective actions.[69/] Thus, as an unauthorized challenge to NRC regulations, this contention should be denied.[70/]

Subsections (D) and (E) apparently relate to the designation of the plume EPZ. It is alleged that local meteorological conditions including temperature inversions must be considered. Applicants submit that local meteorological conditions are not applicable factors discussed in 10 C.F.R. §50.47(c)(2) with regard to the designation of the plume EPZ. The factors which the NRC has included in this determination are: demography, topography, land characteristics, access routes and jurisdictional boundaries.[71/] Other licensing boards have ruled that local meteorological conditions do not play a part in the designation of the plume exposure EPZ inasmuch as

---

68/ Contentions by Joint Intervenors at 6.

69/ See NUREG-0654, supra at 60, which incorporates by reference recommendations of the EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA 520/1-75-001).

70/ See generally Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 2), ALAB-456, 7 NRC 63, 67 n.3 (1978); Potomac Electric Power Company (Douglas Point Nuclear Generating Station, Units 1 and 2), ALAB-218, 8 AEC 79, 89 (1974).

71/ 10 C.F.R. §50.47(c)(2).

meteorology was taken into account by the Commission in selecting a 10 mile radius for the generic plume EPZ.[72]

Further, under the Commission's requirements for emergency planning, meteorological conditions existing at the time of an accident would be available to Applicants via the meteorological instrumentation readings at the Station. Such information would be utilized in determining the protective actions, if any, which had to implemented.

Subsection (F) of joint petitioners' contention apparently argues that the Commission's emergency planning criteria utilized a 3200 megawatt thermal reactor as the model for determing the size of the plume EPZ and that the River Bend reactor is larger than the model. Without conceding that the ten mile EPZ is inappropriate for all sizes of reactors licensed by the NRC, it is noted that the thermal megawatt rating of each of the River Bend reactors is 2894. Therefore, each is actually smaller than the one utilized in devising the plume exposure EPZ size. If joint

---

72/ See NUREG-0396, Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants at Appendix I, I-20-26 (December 1978). The Licensing Board in The Cincinnati Gas & Electric Company (Wm. H. Zimmer Nuclear Power Station), Docket No. 50-358, excluded testimony on local meteorology as it allegedly related to selection of the size of the plume exposure EPZ, finding that the emergency planning regulations had taken this matter into account (Transcript of hearing at 5296-5301 (January 22, 1982)).

petitioners are asserting that the thermal rating for both units must be considered cumulatively, the contention represents a challenge to the Commission's regulations. i.e., that simultaneous accidents in reactor units need not be considered.[73/]

While Applicants believe that the assertions of subsection (G) are without merit and raise questions that in effect challenge the Safety Goal rulemaking, these arguments are more appropriately considered when the plans for the parishes have been submitted. Thus, Applicants will not further respond to them at this time since they are premature.

Applicants' objections to the State of Louisiana's Contention 2 are similar. It should be pointed out that agencies of the State of Louisiana and the parishes have been working towards the selection of a plume exposure EPZ specific to the site around the River Bend Station, which is not completely circular in configuration. Further, Applicants note that plans are being made to assure that adequate protective measures can be taken for the institutions enumerated in the State of Louisiana's petition. It should be clear from the outset that such protective actions need not as a matter of necessity include evacuation depending

---

73/ 10 C.F.R. Part 50, Appendix A, Criterion 5.

upon the type of facilities and their locations. Again, this is a matter to be resolved at a later point.

Joint Petitioners' Contention 10
Construction State

Joint petitioners ask that the record as to contentions be kept open until 15 days before the regulations found at 10 C.F.R. §2.714(b). Initially, this is not a factual contention that may be considered by the Licensing Board. In addition, the Board has already decided this matter. It has permitted several extensions of time,[74] totalling almost nine months, during which the joint petitioners could have submitted additional and more specific contentions. That time has now expired without any effort by petitioners to take advantage of the period afforded them by the Licensing Board to revise and further specify their contentions. The Commission's regulations contemplate that the prehearing conference would be held within 90 days after the notice of opportunity for a hearing.[75] Thus, the Commission has deemed that period of time as a reasonable one within which to formulate specific contentions. In this proceeding, it was not unreasonable for the Board to have established a cut-off for contentions some 18 months after the initial

---

74/ See Memorandum and Order at 3 (July 30, 1982); Order at 2 (August 20, 1982); Order at 2 (December 21, 1982).

75/ 10 C.F.R. §2.751a.

notice of hearing in the Federal Register.[76/] The joint petitioners have not demonstrated otherwise. No legitimate claim for further extensions has been made, and petitioners' request should be denied.

Joint Petitioners' Contention 11
Potassium Iodide Tablets

It is alleged that Applicants have not provided for the distribution and storage of potassium iodide in accordance with accepted public health practice.[77/] Applicants submit that the decision as to whether to utilize potassium iodide for those present within the plume exposure EPZ is a matter that is left to the discretion of the states, and that the Applicants play no part in such decisionmaking.[78/] The Commission has accepted in a number of cases the decisions of cognizable state officials either in favor of or against the provision of potassium iodide.[79/] In fact, in a case where the EPZ falls within the boundaries of two states, one state, constituting approximately one-half of the plume exposure EPZ, decided to utilize potassium iodide for residents and the other state

---

76/ 46 Fed. Reg. 44539 (September 4, 1981).

77/ Contentions by Joint Intervenors at 8.

78/ Thus, NUREG-0654 only requires state and local agencies to identify their plans or intentions regarding the possible administration of potassium iodide. *See* NUREG-0654, *supra* at 63.

79/ [Citation]

in the plume exposure EPZ did not. Thus, Applicants submit there is no litigable issue. The decision of the emergency planning authorities for the particular state to provide or not to provide potassium iodide is binding upon this Licensing Board.[80/]

Joint petitioners allege that monitoring should be expanded to include the human population residing within the ingestion pathway of Iodine-131. While not clear as to what is being requested, it should be noted that plans for monitoring during an incident do call for monitoring Iodine-131.[81/] No litigable issue has therefore been raised.

Joint Petitioners' Contention 12
Funds for Premature or Early Decommissioning [82/]

The matter of the financial qualifications of the Applicants has been completely addressed in the response to the joint petitioners' Contention 1. This contention, which is similarly a prohibited attack on the regulations, and

---

80/ It was reported in 1982 that FEMA had decided not to create a national stockpile of potassium iodide despite a Congressional appropriation for that purpose. A FEMA representative reportedly stated that this decision was based upon widely divergent views by the states as to the necessity for its distribution. See 23 Nucleonics Week 1 (October 14, 1982).

81/ See NUREG-0654, supra at 18 (Table 3), 58.

82/ Joint petitioners misnumbered this as a second Contention 11. Contentions by Joint Intervenors at 8. For reference, Applicants have designated it Contention 12.

even more vague, should likewise be denied. In the Seabrook proceeding, a contention which raised "questions about the financial capability of applicant to safely decommission" was rejected on the basis of the new rule eliminating financial qualification of applicants as an issue in operating license proceedings.[83/]

It should also be noted that the matter of decommissioning is presently under review by the Commission in a rulemaking proceeding[84/] and thus is not appropriate for consideration in individual adjudicatory proceedings. Moreover, neither is the availability of repositories for the long-term storage of spent fuel an appropriate subject of a contention. In the Limerick proceeding, the Licensing Board ruled that the Commission had expressly prohibited such contentions in individual licensing proceedings in light of its rulemaking.[85/] Further, subsequent to the filing of this contention, Federal legislation entitled the Nuclear Waste Policy Act of 1982, Public Law No. 97-425 (January 7, 1983) was enacted to deal with specific arrangements regarding the disposal of reactor fuel. This contention raises no litigable issue and should be denied.

83/ Seabrook, supra, slip op. at 95.

84/ 47 Fed. Reg. at 13751.

85/ Limerick, supra, 15 NRC at 1455. See also Sacramento Municipal Utility District (Rancho Seco Nuclear Generating Station), ALAB-655, 14 NRC 799, 816 (1981).

Joint Petitioners' Contention 13 86/
State of Louisiana Contention 5
Construction of Unit 2 During Unit 1 Operation

In these contentions, joint petitioners and the State of Louisiana allege that Applicants have failed to consider the effect of future construction activities for Unit 2 on the safe operation of Unit 1. The State cites 10 C.F.R. §50.34(b)(6)(vii) as basis for its contention, and correctly asserts that Unit 2 is currently not scheduled and construction on that unit has been halted. Moreover, Applicants will make no decision with regard to Unit 2 until late 1985.

Under these circumstances, it is entirely reasonable for Applicants not to have submitted detailed plans with regard to the construction of Unit 2. Inasmuch as the timing, schedule and number of construction workers could not be determined with any specificity, no specific plan could be submitted at this point which would satisfy the cited regulation. As is obvious, the NRC Staff would need to review and approve such plans only prior to the resumption of Unit 2 construction. It is therefore premature to consider this contention, which should be denied. The Commission has adequate remedies both within and outside its adjudicatory hearing process for this matter to be raised

---

86/ This contention has been renumbered. See n.82, supra.

should Unit 2 construction be recommenced. Applicants, of course, will notify the NRC of their intent to resume construction of Unit 2.87/

State of Louisiana Contention 1
Table S-3

State of Louisiana Contention 1 alleges that "[a]pplicants have 'failed to allow for proper consideration of the uncertainties concerning the long-term isolation of high-level and transuranic wastes, and ...failed[ed] to allow for proper consideration of the health, socioeconomic and cumulative effects of fuel-cycle activities.'"88/ The contention is based entirely upon a recent decision of the Court of Appeals for the District of Columbia Circuit which invalidated the Commission's rule for consideration of the fuel cycle in individual reactor licensing proceedings, the so-called Table S-3 to 10 C.F.R. Part 51.89/ Commission precedents as well as its specific instructions in this

---

87/ Applicants would note that the Commission has recently rejected a rulemaking petition filed by Wells Eddelman, 47 Fed. Reg. 46524 (October 19, 1982), which would have required separate operating license hearings for multi-unit facilities. Applicants submit that this action confirms the propriety of considering the issuance of operating licenses for both Units 1 and 2, recognizing that the particular information regarding Unit 2 may be properly submitted at a later time.

88/ Supplemental Petition at 2.

89/ Natural Resources Defense Council, Inc. v. Nuclear Regulatory Commission, 685 F.2d 459 (D.C. Cir. 1982), cert. granted, 51 U.S.L.W. 3419 (November 29, 1982) (No. 82-545) ("S-3 decision").

instance are clear that, pending completion of Supreme Court review, this matter may not be litigated in individual licensing proceedings.

The State of Louisiana attempted to raise precisely this issue in Mississippi Power & Light Company (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-704, 16 NRC ___ (December 8, 1982). The Appeal Board affirmed the Licensing Board's denial of Louisiana's intervention petition which was based, as here, on the Table S-3 decision. There, the Appeal Board concluded that "guidance of the Commission leaves no room for doubt that the question of safe waste disposal as reflected in the S-3 table of effluent releases is not a matter for case-by-case litigation in individual reactor licensing proceedings at this time."90/

The Appeal Board based its decision in large part upon a November 8, 1982 Commission policy statement with regard to the District of Columbia's S-3 decision.91/ Therein, the Commission determined that Table S-3 issues could not be litigated on a case-by-case basis:

> To move further toward case-by-case litigation would reintroduce the significant burdens the rule was intended to relieve. Use of the S-3 rule has served the important purpose of providing the

---

90/ Grand Gulf, supra, slip op. at 12.

91/ "Statement of Policy, Licensing and Regulatory Policy and Procedures for Environmental Protection; Uranium Fuel Cycle Impacts," 47 Fed. Reg. 50591 (November 8, 1982) ("S-3 Policy Statement").

> underlying basis for consideration of fuel cycle impacts, and the Commission believes that an attempt to proceed without the rule would probably prove unworkable . . . . The resulting delay and drain on staff resources would be substantial, and would not only delay licensing of qualified facilities, but would also substantially disrupt the Commission's regulatory program, including its program to develop safety standards for high-level waste disposal facilities. 92/

The Appeal Board has directed that Licensing Boards "act as if the District of Columbia Circuit's decision, which is now under review by the Supreme Court, is currently of no operative effect.[93/] Thus, this contention must be denied.

State of Louisiana Contention 6
Asiatic Clams

It is alleged that Applicants have failed to provide adequate assurance that River Bend Station components and systems relying on Mississippi River water for their operation will be adequately protected against infestation of Asiatic clams (Corbicula leana). With regard to this contention, the State has attached a letter of February 14, 1983 from the Applicants which addresses the question of steps necessary to control any infestation of Asiatic clams including details on chlorination levels. Compared to the

---

92/ 47 Fed. Reg. at 50592 (footnote omitted).

93/ Grand Gulf, supra, slip op. at 12-13.

Applicants' specific discussion of the matter in that letter and in Section 3.6.1.3.2 of the EROLS, the State of Louisiana merely asserts without basis of any kind that Applicants have failed to demonstrate that infestation can and will be controlled. This contention is completely lacking in specificity and bases and should be rejected.

Conclusion

As discussed in detail above, the contentions of both the State of Louisiana which as intervened through the office of Attorney General and the joint petitioners are defective and should not be considered for litigation before this Licensing Board. Furthermore, a number of the matters raised are premature and under the Commission's precedents must be rejected at this time.

Respectfully submitted,

CONNER & WETTERHAHN, P.C.

Mark J. Wetterhahn

Mark J. Wetterhahn
Counsel for the Applicants

April 15, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
GULF STATES UTILITIES COMPANY, et al.	)	Docket Nos. 50-458 OL 50-459 OL
	)	
(River Bend Station, Unit 1 and 2	)	

SERVICE LIST

I hereby certify that copies of "Applicant's Answer to the Contentions Filed by Petitioners for Leave to Intervene, State of Louisiana through the Office of the Attorney General, Louisiana Consumer's League, Louisianans for Safe Energy, Inc. and Gretchen Reineke Rothchild" dated April 15, 1983, in the captioned matter, have been served upon the following by deposit in the United States mail this 15th day of April, 1983:

B. Paul Cotter, Jr., Esq.
Chairman, Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Forrest J. Remick
305 East Hamilton Avenue
State College, PA 16801

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

David A. Repka, Esq.
Counsel for NRC Staff
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

James W. Pierce, Jr., Esq.
P. O. Box 23571
Baton Rouge, LA 70893

Doris Falkenheiner, Esq.
Stephen M. Irving, Esq.
355 Napoleon Street
Baton Rouge, LA 70802

William Guste, Jr., Esq.
Attorney General
State of Louisiana
234 Loyola Avenue
New Orleans, LA 70112

Ian D. Lindsey, Esq.
Department of Justice
7434 Perkins Road
Suite C
Baton Rouge, LA 70808

Docketing & Service Section
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Linda B. Watkins, Esq.
355 Napoleon Street
Baton Rouge, LA 70802

Gulf States Utilties
Company
Attn: Mr. James E. Booker
Manager - Engineering
and Licensing
P. O. Box 2951
Beaumont, Texas 77704

Mark J. Wetterhahn

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DOCKETED
USNRC

Docket Nos: 50-458
and 50-459

'83 APR 18 A10:08

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

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

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

Enclosure:
As stated

cc: See next page

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

cc: Troy B. Conner, Jr., Esquire
Conner and Wetterhahn
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

Mr. J. E. Booker
Manager -Technical Programs
Gulf States Utilities Company
Post Office Box 2951
Beaumont, Texas 77704

Stanley Plettman, Esquire
Orgain, Bell and Tucker
Beaumont Savings Building
Beaumont, Texas 77701

Karin P. Sheldon, Esquire
Sheldon, Harmon & Weiss
1725 I Street, N. W.
Washington, D. C. 20006

William J. Guste, Jr., Esquire
Attorney General
State of Louisiana
Post Office Box 44005
State Capitol
Baton Rouge, Louisiana 70804

Richard M. Troy, Jr., Esquire
Assistant Attorney General in Charge
State of Louisiana Department of Justice
234 Loyola Avenue
New Orleans, Louisiana 70112

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 Toughness (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 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.

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.

GULF STATES UTILITIES COMPANY
POST OFFICE BOX 2951 · BEAUMONT, TEXAS 77704
AREA CODE 713 838-6631

DOCKETED
USNRC

'83 APR 18 A10:08

June 3, 1982
RBG-12,758
File No. G9.5

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
Office of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Schwencer:

River Bend Station Units 1 & 2
Docket Nos. 50-458 & 50-459

This letter is provided in response to your November 19, 1981, request for additional information regarding the status of unresolved safety issues at Gulf States Utilities (GSU). GSU participated in the Licensing Review Group II (LRG-II) activities pertaining to the generic resolution of unresolved safety issues. Position paper 1-GIB was included in the submittal by the LRG-II via a letter dated January 25, 1982 to Howard J. Faulknuer (NRC) from D. L. Holtzcher, Chairman of the LRG-II Working Group. GSU intends to formally endorse this position on our docket in the near future, together with the other applicable LRG-II positions. A copy of 1-GIB is attached for your convenience.

Sincerely,

J E Booker

J. E. Booker
Manager-Engineering & Licensing
River Bend Nuclear Group

JEB/LAE/kt

Attachment

INTERIM LICENSING BASES PENDING RESOLUTION OF UNRESOLVED SAFETY ISSUES

ISSUE

LRG-II plants will develop unified bases and justification for licensing and operation while the identified generic safety issues remain unresolved and provide a summary description of relevant investigative programs and interim measures pending resolution of the unresolved safety issues.

LRG-II RESPONSE

LRG-II participants have reviewed the generic issues identified in NUREG-0606, "Unresolved Safety Issues." The following information is provided for each of the applicable "Unresolved Safety Issues" as a bases for licensing prior to ultimate resolution of these issues.

MJA:rf:csc/118A7

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 waterhammers in pressurized and boiling water reactors have been reported. The waterhammers (or steam hammers) have involved steam generator feedrings and piping, and 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 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, "Waterhammer in Nuclear Power Plants" (July, 1979), providing the results of an NRC staff review of waterhammer events in nuclear power plants and stating current 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 the LRG II plants 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 LRG-II plants emergency core cooling systems against the effects of waterhammer, the ECC systems are provided with jockey pumps. These jockey pumps 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 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

A-1 Waterhammer (Cont'd)

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 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.

In addition, LRG-II participants will conduct a preoperational vibration and dynamic effects test program in accordance with Standard OM-3 of the American Society of Mechanical Engineers for all Class 1, Class 2, Class 3 and other piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips, and other operating modes.

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 system 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 haved not explicitly been accounted for in the design and operation of LRG-II plants, corrective measures will be implemented at that time. The task has not identified the need for measures beyond those already implemented.

Based on the foregoing, we conclude that the LRG-II plants can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

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 (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 system did not "scram" as desired, then an "anticipated transient without scram," or ATWS, would have occurred.

A.-9 Anticipated Transients Without Scram (Cont'd)

All boiling water reactors, including LRG-II plants, have been required to provide recirculation pump trip in the event of a reactor trip and to provide additional operator training for recovery from anticipated transients without scram events. In addition, LRG-II plants will implement emergency procedures and operator training to cope with potential anticipated transients 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 anticipated transient without scram issue is currently under review through the rulemaking proceedings. Notice of the proposed rule for ATWS was published in the Federal Register on November 24, 1981. LRG-II plants will 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 the LRG-II plants 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.

A.-11 Reactor Vessel Materials Toughness (Cont'd)

Based upon evaluation of the LRG-II reactor vessel's materials, toughness, we conclude that adequate safety margins exist for brittle failure during operating, testing, maintenance, and anticipated transient conditions over the life of the units. Since Task Action Plan A-11 is projected to be completed well in advance of LRG-II plants reactor vessels reaching a fluence level which would notably reduce fracture resistance, acceptable vessel integrity for the postulated accident conditions will be assured. When Task Action Plan A-11 is completed and explicit fracture evaluation criteria for accident conditions are defined, all vessels will be reevaluated for acceptability over their design lives.

The materials of the LRG-II reactor vessels meet the fracture toughness requirements of NB-2300 of the ASME Code. Based on this fact and the fabrication techniques employed on the vessel, we estimate that the total fluence over the design life would result in a final fracture toughness value above the minimum charpy impact requirement of 50 foot-pounds. In addition, the surveillance program required by Appendix H of 10CFR Part 50 will afford an opportunity to reevaluate the fracture toughness periodically during a minimum of the first half of the design life.

To assure adequate safety margins, adjustment to the nil ductility transient temperature (NDTT) and the development method for pressure/temperature curves are specified in 10CFR50 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 core 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.

Therefore, based upon the foregoing, we conclude that LRG-II plants 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

The licensing requirements and procedures used in the LRG-II plant safety reviews 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.

MJA:rf:csc/118A11

A-17 Systems Interaction in Nuclear Power Plants (Cont'd)

These design provisions supplemented by the current review procedures of the Standard Review Plan, 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.

In mid-1977, Task A-17 was initiated to confirm that present review procedures and safety criteria provide an acceptable level of redundancy and independence for systems required for safety by evaluating the potential for undesirable interactions between and among systems.

The NRC staff's current review procedures assign primary responsibility for review of various technical areas and safety systems to specific organizational units and assign secondary responsibility to other units where there is a functional or interdisciplinary relationship. Designers follow somewhat similar procedures and provide for interdisciplinary reviews and analyses of systems. Task A-17 provide an independent study of methods that could identify important systems interactions adversely impacting safety, and which are 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 and among systems and have the potential for negating or seriously degrading the performance of safety functions. The study concentrated on common cause of linking failures among systems that could violate a safety function. The investigation then identified where NRC review procedures may not have properly accounted for these interactions.

The Sandia Study 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 againt a sample of some systems interaction candidates derived from Licensee Event Report evaluations.

MJA:rf:csc/118A12

A-17 Systems Interaction in Nuclear Power Plants (Cont'd)

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.

LRG-II participants will provide for a systematic visual inspection by a multidisciplinary team to review the "as-built" condition of the plant areas where physical interactions could potentially result in adverse effects on safety-grade equipment. Visual inspections of interaction areas are performed for spatially coupled systems interactions initiated by seismic events. Any spatial separations that do not meet established design criteria are reported for disposition by analysis and/or hardware modification. LRG-II participants are improving their programs based on the experience gained in the industry's efforts, but will maintain the multidisciplinary team concept which the staff considers essential to a systems interaction analysis.

Therefore, we conclude that these is reasonable assurance that the LRG-II plants can be operated prior to 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 cooling 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.

MJA:rf:csc/118A13

A-39 Safety Relief Valve Hydrodynamic Loads (Cont'd)

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 while 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 at LRG-II plants have been 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. NRC has approved the revised GE method in NUREG-0802. The LRG-II plants have used the revised SRV loads accepted by the NRC and will review the inplant testing results from Kuo Sheng 1 and Grand Gulf-1, to determine their applicability and confirm the conservatism of their designs. This concern is addressed in further detail in LRG-II issue 1-CSB.

A-40 Seismic Design Criteria - Short-Term Program

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, these 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, re-reviews 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 the LRG-II plants have been established on current licensing criteria and requirements. Should the resolution of Task A-40 indicate a change is needed in these licensing requirements, all operating reactors including Clinton, Perry and River Bend will be reevaluated on a case by case basis.

MJA:rf:csc/118A14

A-40 Seismic Design Criteria - Short-Term Program (Cont'd)

Accordingly, we have concluded that the LRG-II plants 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 through the reactor system by the emergency core cooling pumps to maintain core cooling. Loss of the ability to draw water from the suppression pool could disable the emergency core cooling system.

The concern addressed by this Task Action Plan for boiling water reactors is primarily focused on 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. LRG-II plants have a minimum suction submergence for the ECCS systems of over 7 feet. This concern is addressed in further detail in LRG-II issue 7-RSB.

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 minimum of 4 feet and the approach velocity is only 1 foot/second above the pool bottom. In addition, boiling water reactor designs employ strainers on the suction sized with flow areas 200 percent larger than the suction piping.

Accordingly, we conclude that LRG-II plants 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 two offsite alternating current power connections, a standby emergency diesel generator alternating current power supply, and direct current sources.

MJA:rf:csc/118A15

A-44 Station Blackout

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 event, 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 have 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 LRG-II facilities. Nonetheless, a combination of design, operating, and testing requirements have been imposed to 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 the core will be cooled. These design, operating, and testing requirements are discussed below.

A loss of offsite alternating current power involves a loss of both the preferred and backup sources of offsite power. Our review and basis for acceptance of the design, inspection, and testing provisions for the offsite power system are described in Section 8.2 of the SER.

If offsite alternating current power 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 Sections 8.3 and 9.5 of the SER. The requirements include preoperational testing to assure the reliability of the installed 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 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 accommodate a station blackout event. The Commission has reviewed their recommendations and determined that some interim measures should be taken at all facilities including LRG-II plants while Task A-44

A-44 Station Blackout (Cont'd)

is being conducted. NRC Generic Letter 81-04 requested a review of plant capability to mitigate a station blackout event and prompt implementation, as necessary, of emergency procedures and a training program for station blackout events. Consequently, interim emergency procedures and operator training for safe operation of the facility and restoration of alternating current power will be implemented. This action will be completed by fuel load date.

Based on the above, we have concluded that there is reasonable assurance that LRG-II plants 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 product 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 from 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 LRG-II plants are designed with various methods for the removal of decay heat. As discussed above, the decay heat is normally rejected to the turbine condenser and condensate is returned to the vessel by the feedwater system. The reactor core isolation cooling (RCIC) system provides an alternate means of supplying makeup water to the vessel. This turbine driven pump takes suction from the condensate storage tank and pumps to the vessel. 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 (HPCS) system is provided if the reactor core isolation cooling system is not available. Both of these systems (RCIC and HPCS) can supply water to the vessel from either the condensate storage tank or the suppression pool.

MJA:rf:csc/118A17

A-45 Shutdown Decay Heat Removal Requirements (Cont'd)

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 for the LRG-II plants 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 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 acceptability of current designs. A report of these analyses was provided to the NRC in NEDO-24708A Revision 1, dated December, 1980.

Based on the above, we have concluded that the LRG-II plants can be operated prior to the ultimate resolution of the 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 shutdown the plant, as well as equipment whose functions is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

A-46 Seismic Qualification of Equipment in Operating Plants (Cont'd)

LRG-II plants were designed using current seismic design criteria, and methods for seismic equipment qualification are to be latest codes and standards. Requirements for seismic equipment qualification include IEEE 344-1975 and Regulatory Guides 1.92 and 1.100. Standard Review Plans 3.2.2, 3.9.2, 3.9.3, and 3.10 have also been considered in the qualification efforts.

Since identification of hydrodynamic load effects on LRG-II plant structures, an effort was initiated to assess the effects of these loads (in combination with previously established seismic loads) on equipment required to safely shut down the plant. This reassessment involved validation of equipment qualification through both analytical methods and additional testing, where required.

It is concluded that LRG-II plants 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 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 sever than those transients analyzed as anticipated operational occures. 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 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 LRG-II plant 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

A-47 Safety Implications of Control Systems (Cont'd)

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 systematic evaluation of the control system design, such as contemplated for this "Unresolved Safety Issue," has not been performed to determine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than presently analyzed. However, 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 system action. Specifically, these reviews include identification and evaluation of the potential adverse impacts to plant safety as a result of control system failures, effects from loss of non-Class 1E power sources, and harsh environments following high energy line breaks. These concerns are addressed in further detail in LRG-II issues 5-ICSB, 6-ICSB, and 7-ICSB.

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 LRG-II plants 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.

Based on the above, we have concluded that there is reasonable assurance that LRG-II plants 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

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (Cont'd)

of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal, and radiolytic effects of post-accidents environmental conditions on containment protective coating systems and electric cable insulation.

Because of the potential for significant hydrogen generation as the result of an accident, 10CFR Section 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 10CFR Part 50, require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

Regulation 10CFR 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 10CFR 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 10CFR 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. 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, 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.

MJA:rf:csc/118A21

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (Cont'd)

The LRG-II position is to comply with this interim rule through use of a hydrogen igniter system. This system consists of glow plug igniters distributed throughout the containment. This system is designed to ignite hydrogen at low concentrations, thereby maintaining the concentration of hydrogen below its detonable limit and preventing potential containment overpressure.

To collectively evaluate the concerns associated with the Hydrogen issue for Mark III containments, LRG-II participants are involved in an owners group. This group is sponsoring analytical work with General Electric, Offshore Power systems and others. Current evaluations of this group have demonstrated that containment pressures will remain well below the failure point as the result of the anticipated hydrogen release and burn.

Based on the above, we conclude that LRG-II plants 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.

MJA:rf:csc/118A22