January 8, 1982



UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

| | | |
|---|---|---|
| In the Matter of | § | |
| | § | |
| HOUSTON LIGHTING & POWER COMPANY | § | Docket No. 50-466 |
| | § | |
| (Allens Creek Nuclear Generating | § | |
| Station, Unit No. 1 | § | |

# APPLICANT'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW ON RADIOLOGICAL HEALTH AND SAFETY ISSUES IN THE FORM OF AN INITIAL DECISION



OF COUNSEL:

BAKER & BOTTS
3000 One Shell Plaza
Houston, Texas 77002

LOWENSTEIN, NEWMAN, REIS
& AXELRAD
1025 Connecticut Ave., N.W.
Washington, D.C. 20036

J. Gregory Copeland
Scott E. Rozzell
3000 One Shell Plaza
Houston, Texas 77002

Jack R. Newman
Robert H. Culp
David B. Raskin
1025 Connecticut Ave., N.W.
Washington, D.C. 20036

ATTORNEYS FOR APPLICANT
HOUSTON LIGHTING & POWER COMPANY

DS03
s
1/1



G

TABLE OF CONTENTS

Radiological Health and Safety Findings

January 8, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

| | | |
|---|---|---|
| In the Matter of | § | |
| | § | |
| HOUSTON LIGHTING & POWER COMPANY | § | Docket No. 50-466 |
| | § | |
| (Allens Creek Nuclear Generating Station, Unit No. 1) | § | |
| | § | |

# APPLICANT'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW ON RADIOLOGICAL HEALTH AND SAFETY ISSUES IN THE FORM OF AN INITIAL DECISION

## I. INTRODUCTION

Applicant, Houston Lighting & Power Company (HL&P) hereby submits, pursuant to 10 CFR § 2.754, its Proposed Findings of Fact and Conclusions of Law in the form of an Initial Decision on radiological health and safety issues and requests that they be adopted by the Atomic Safety and Licensing Board (ASLB) in this proceeding.*/

As required by 10 CFR § 2.754(c), the exact record reference relied upon is cited with respect to each proposed

*/ Pursuant to the Joint Motion of Applicant and Staff to Establish a Schedule to File Proposed Findings of Fact and Conclusions of Law, which was adopted by the Board on November 16 (Tr. 19875), the Staff will file its proposed findings on environmental issues concurrently with Applicant's filing of the instant findings. The Staff intends to include in its findings, a statement updating the history of the case from the date of the issuance of the Partial Initial Decision. Accordingly, in order to avoid repetition, Applicant has not included in this document a statement of the case.

finding of fact herein and each conclusion of law is accompanied by the authorities or reasoning which the Applicant believes support the conclusion requested.

## II. NONCONTESTED MATTERS

### A. General

1 The Notice of Hearing issued with respect to this proceeding on December 20, 1973, requires the Board, pursuant to the Atomic Energy Act of 1954, as amended, to consider and decide:

1. Whether in accordance with the provisions of 10 CFR § 50.35(a):

   (a) The Applicant has described the proposed design of the facility including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection for the health and safety of the public;

   (b) Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration, will be supplied in the final safety analysis report;

   (c) Safety features or components, if any, which require research and development have been described by the Applicant and the Applicant has identified, and there will be conducted a research and development program reasonably designed to resolve any safety questions associated with such features or components; and

   (d) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facilities, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facilities can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

2. Whether the Applicant is technically qualified to design and construct the proposed facilities;

3. Whether the Applicant is financially qualified to design and construct the proposed facilities; and

4. Whether the issuance of permits for construction of the facilities will be inimical to the common defense or to the health and safety of the public.*/

2 The application for a construction permit and operating license for the Allens Creek Nuclear Generating Station (ACNGS) before this Board consists of (a) general and financial information required by 10 CFR § 50.33 and contained in the License Application; (b) the Environmental Report (ER) and the Supplemental Environmental Report (ER Supp); (c) the Preliminary Safety Analysis Report (PSAR);**/ and (d) testimony and evidence presented through Applicant's witnesses at the evidentiary hearings.

---

*/ 38 Fed. Reg. 35521 (1973). This Notice of Hearing reflects the original intent of the Applicant to construct two units at ACNGS. In reactivating the project in 1977, Applicant reduced the number of units to one.

**/ Several letters have been submitted by the Applicant to the Staff to clarify or reaffirm positions taken in the PSAR. These letters are treated as having the same effect as supplements to the PSAR and shall be treated as such.

B. Facility Description*/

3 As stated in paragraph 80 of the Partial Initial Decision (PID), the ACNGS will be located in Austin County, Texas, on a site 45 miles west of Houston, Texas. The nuclear steam supply system (NSSS) is being provided by the General Electric Company (GE). The reactor will be one of GE's BWR/6-238 class of boiling water reactors. This BWR has been designed for a thermal output of 3579 megawatts (MW) and a net electrical output of 1146 MW. All safety systems have been analyzed and will be designed for an ultimate thermal output of 3758 MW. (App. Exh. 27 § 1.1; Staff Exh. 23 § 1.1).

4 The fuel design for ACNGS will consist of 748 8x8 fuel assemblies with 62 fuel rods per assembly. Each fuel rod will be made of slightly enriched uranium dioxide ($UO_2$) fuel pellets sealed in Zircaloy-2 tubes. The fuel design will consist of fuel rods prepressurized

---

*/ The following paragraphs provide a brief general description of the facility in accordance with the Notice of Hearing Item 1(a). An extended discussion in this section, however, is unnecessary in light of the numerous contested health and safety issues which relate to practically every aspect of the proposed facility. The findings made by the Board on those issues which appear hereafter have also been considered in the determination that Applicant has complied with the provisions of Item 1(a) of the Notice of Hearing.

with helium and two water rods per fuel assembly. This fuel design is essentially identical with the design for the La Salle nuclear facility which the Staff approved in March 1981 (NUREG-0519). (Staff Exh. 21, § 4.2; App. Exh. 27 § 4.1.2.1).

5 Water will flow through the reactor core and will serve both as a neutron moderator and as a coolant. Water is moved through the reactor by the driving force of 20 jet pumps and by convective forces. The steam which is generated in the reactor by the heat from the nuclear reaction will be separated and dried, then vented through the four main steam lines to the turbine generator system where it will be utilized in the generation of electricity. The steam from the turbine generator will be exhausted to a main condenser which is located beneath the turbine and where the condensate will be collected and returned through a cleanup system and feedwater heaters for recycling through the reactor vessel and the core. The condenser will be cooled by drawing water from the ACNGS cooling lake. The heated cooling water is then returned to the lake. (App. Exh. 27 §§ 1.2.2.2.2, 5.5.5; Staff Exh. 23 § 1.2).

6 The reactor coolant pressure boundary (RCPB) consists of the reactor vessel itself, two recirculation lines, four main steam lines, two feedwater lines, and branch

lines to their outer most isolation valves. Enclosing the RCPB will be a reinforced concrete cylindrical structure called the "drywell". The purpose of the drywell is to contain and direct steam from a postulated accident to the suppression pool where the steam will be condensed. The suppression pool is located between the drywell and a free standing steel structure called the "containment" which completely encloses the drywell. Surrounding the containment is a reinforced concrete shield building. Collectively, these structures comprise the ACNGS Mark III containment system and are designed to mitigate the consequences of postulated accidents. (App. Exh. 27 §§ 1.2.2.3.9; 6.2.5; Staff Exh. 23 § 1.2).

7 The reactor can be shut down rapidly by use of bottom-entry, cruciform shaped control rods which are moved vertically into the spaces between the fuel assembly channels by a hydraulic mechanism. Each control rod will be independent of other rods and will have its own hydraulic control system. A standby liquid control system will also be available to inject a boron solution into the reactor. The ACNGS reactor will be provided with certain engineered safety features which will ensure the capability to isolate the containment,

shutdown the reactor, restrict radioactivity releases to acceptable levels, provide for heat removal for long term core cooling, and condense steam within the primary containment in the event of a postulated design basis accident. Part of the engineered safety features will be an emergency core cooling system (ECCS), which is designed to provide coolant to the reactor core in the event of an accident. The ECCS will be capable of operating even if there is a loss of offsite electrical power through the use of onsite diesel generators. (App. Exh. 27 §§ 1.2.2.3; 6.3; Staff Exh. 23 § 1.2).

8 The Applicant has employed Ebasco Services, Inc. to perform services as the architect-engineer and constructor for ACNGS. (App. Exh. 27 §§ 1.1; 1.4; Staff Exh. 23, § 1.1).

9 The Applicant has identified certain development programs applicable to ACNGS that remain to be undertaken. (Staff Exhs. 23, § 1.8 and 20, pp. II-4 to II-15, II-33). When completed, these tests programs will conclude the tests necessary for the verification of the design and safe operation of the ACNGS.

C. ACRS Review

10 The Advisory Committee on Reactor Safeguards (ACRS) reviewed the proposed ACNGS facility and transmitted

its favorable conclusions to the Commission in a letter dated December 12, 1974 (Staff Exh. 18, Appendix I). When the ACNGS application was reactivated in 1977 as a single unit, the ACRS decided that there was no need to reopen its review of the proposed facility. (Staff Exh. 19, § 18.0). The Committee further decided that it would not review the TMI action items applicable to ACNGS, which are addressed in Supplement 3 to the SER (Staff Exh. 21, pp. 1-2). However, the ACRS did request a Staff briefing on the Applicant's post accident inerting system, which was made on September 10, 1981, and subsequently informed the Staff that it wanted to review this system prior to approval of it by the Staff or before alternative means of hydrogen control were precluded. The Staff will comply with the Committee's request (Staff Exh. 21, pp. 1-2). In addition, the Staff proposes to condition the construction permit to require the Applicant to report to the Staff, on an established schedule, the status of its proposed hydrogen control system, and to maintain the capability of incorporating alternative systems pending Staff review and approval of the proposed system. (Staff Exh. 21, pp. 1-2 to 1-3).

D. Staff Safety Reviews Including TMI Action Plan

11 The Staff has reviewed all aspects of the application to construct ACNGS to determine whether ACNGS can be constructed and operated without endangering the public health and safety. The Staff has documented its review of the ACNGS application in its Safety Evaluation Report issued in November, 1974 (Staff Exh. 23) and in Supplement 1 issued in June, 1975 (Staff Exh. 18), Supplement 2 issued in March, 1978 (Staff Exh. 19), Supplement 3 issued in July, 1981 (Staff Exh. 20), and Supplement 4 issued in October, 1981 (Staff Exh. 21) to that report. The Board has reviewed the SER and the Supplements, and finds that the Staff's review of the ACNGS application has been adequate and concurs that the proposed facility can be constructed and operated without endangering the health and safety of the public.

12 As a result of the lessons learned from the Three Mile Island (TMI) incident as set forth in the NRC's Action Plan (NUREG-0660, May, 1980), the Staff issued a document entitled "Licensing Requirements for Pending Applicants for Construction Permits and Manufacturing License" (NUREG-0718, Revision 1, July, 1981). This document sets forth the then current Staff requirements which must be addressed by construction permit appli-

cants in order to comply with the lessons learned from the TMI incident. The Commission has published in the Federal Register (46 Fed. Reg. 18045, March 23, 1981) proposed amendments to its regulations, based upon the requirements of NUREG-0718, to add licensing requirements applicable to construction permit applications. Although a final rule has not yet been published, the Commission has authorized the Staff to proceed with the review of pending construction permit applications on the basis of NUREG-0718, Revision 1 pending publication of a final rule on this matter. (Staff Exh. 20, p. 1).

13 In Amendments 57 and 59 to the PSAR (Appendix O), the Applicant submitted its responses to the requirements set forth in NUREG-0718, Revision 1. The Staff has reviewed these responses and issued a safety evaluation report (SER, Supp. 3) which documents the Applicant's compliance with these requirements. At the evidentiary hearings in this proceeding, there was testimony from both Applicant and Staff witnesses relating to many of the issues covered in Supplement 3 to the SER and the Board's findings on those issued are found in the discussion of individual contested issues. However, the Board has also reviewed Supplement 3 to the SER and finds that the Staff's review and approval of the Applicant's TMI submittal has been adequate.

## III. CONTESTED ISSUES

Baker Contention 1 (Cumings 1; TexPirg AC 32; Doggett 4; Perrenod 1): Financial Qualifications.

14 In his Contention 1, Mr. Baker alleges that the Applicant does not satisfy the financial requirements of 10 C.F.R. §50.33(f). That provision of the Commission's regulations requires an applicant for a construction permit to demonstrate that it has "reasonable assurance of obtaining the necessary funds" to cover estimated construction and fuel cycle costs. In Public Service Company of New Hampshire, et al. (Seabrook Station, Units 1 and 2), 7 NRC 1 (1978) the Commission described the showing required of an applicant as follows:

> "...given the history of the present rule and the relatively modest implementing requirements in Appendix C, a 'reasonable assurance' does not mean a demonstration of near certainty that an applicant will never be pressed for funds in the course of construction. It does mean that an applicant must have a reasonable financing plan in light of the relevant circumstances". 7 NRC at 18. (Footnote omitted).

15 The cost of constructing ACNGS is currently estimated to be $2.090 billion, a figure which takes into account inflation and the added costs of complying with the Commission's new requirements stemming from the accident at Three Mile Island. (Applicant's witness Dean,

p. 2, following Tr. 16723).*/ However, ACNGS is only one of the construction projects that Applicant plans to have underway during the period 1981-1991. While Applicant's overall plan requires substantial capital expenditures in the next 10 years, the Applicant has stretched out its program for constructing additional power plants in order to minimize any constraints on its ability to finance these new facilities. (Applicant's witness Dean pp. 9-10, following Tr. 16723). Under this construction program, total construction expenditures for that period are estimated to be $13.12 billion (Applicant's witness Dean p.4, following Tr. 16723). While the size of the current construction program is quite large, the growth in Applicant's total assets and capitalization during the period 1981-1991 will actually be slower than that experienced by HL&P from 1970-1980. (Applicant's witness Dean p.4, following Tr. 16723).

16 With this historical performance as background, Mr. Dean, the Applicant's Chief Financial Officer presented HL&P's plan for financing its construction program (Exhibit HRD-1, following Tr. 16723). The plan fea-

---

*/ Of course, not all of the cost of ACNGS will have to be incurred in the future. As of June 30, 1981, the Applicant had already spent $282 million on ACNGS. (Applicant's witness Dean, p.2 following, Tr. 16723).

tures a combination of internally generated funds (39%) and external funds in the form of debt and/or equity securities (61%). (Applicant's witness Dean, pp. 3-4, following Tr. 16723).

17 During the eleven year period ending on December 31, 1980, internally generated funds had supplied a somewhat higher percentage (41%) of the Applicant's total construction expenditures. (Applicant's witness Dean, p. 4, following Tr. 16723). Although frequent rate increases will be necessary to provide HL&P with sufficient internally generated funds for its construction program, Mr. Dean expressed his confidence that the Public Utility Commission of Texas would maintain its practice of providing adequate and timely rate relief. (Applicant's witness Dean, p. 9, following Tr. 16723). In fact, the forecast prepared by Applicant assumes a return on equity equal to that previously granted to the Applicant by the Public Utility Commission. (Applicant's witness Dean, p. 9, following Tr. 16723).*/

---

*/ The Commission has published a proposed rule which would eliminate the financial qualifications review at the CP stage. 46 Fed. Reg. 41,786 (Aug. 18, 1981). In so doing, the Commission expressed the view that the ratemaking process is the overriding factor in assessing an Applicant's financial qualifications to construct a facility. Id., at 41,788.

18 Mr. Dean also testified that the Company expects adequate markets to exist for its external financing. In terms of total assets, HL&P is one of the financially strongest utilities among those in the U.S. that own nuclear generating facilities. The Applicant's bonds and preferred stocks are rated no lower than "A" by the major rating agencies, a rating which characterizes such securities as upper medium grade investments. (Applicant's witness Dean, p. 5, following Tr. 16723). The Applicant's projections for the construction period in the areas of its interest coverage, return on equity and capitalization are considerably more favorable than current figures for other "A" rated utilities. (Applicant witness Dean, pp. 5-6, following Tr. 16723). When coupled with the favorable regulatory climate in Texas, this evidence of Applicant's financial strength supports the proposition that the necessary external capital will be available to finance the construction program which includes ACNGS.

19 The Staff reviewed Applicant's financing plan and found it to be reasonable. ( SER Supp. No. 4, Staff Exh. 21, adopted as testimony by Staff's witness Petersen, pp. 20-1 to 20-8, following Tr. 20433). Using the DOE's CONCEPT capital cost model, Staff found the

Applicant's projected cost of the ACNGS to be reasonable, perhaps conservative. (Staff's witness Petersen, p. 20-1, following Tr. 20433).

20 Mr. Petersen, the senior financial analyst at the NRC, testified that the most significant factor in the review of a financing plan was the projected rate of return on common equity (Staff's witness Petersen, p. 20-2, following Tr. 20433). The Applicant used as its assumed rate of return the actual rate granted the Company by the Texas Public Utility Commission in September, 1980. Staff found that the regulatory environment in Texas was favorable and that the use of such rate was reasonable. (Staff's witness Petersen, pp. 20-5, following Tr. 20433).

21 Staff further found that the interest coverage and capitalization assumptions used by the Applicant were reasonable on the basis of historical performance and projected debt issuances. (Staff witness Petersen, pp. 20-7, following Tr. 20433).

22 As a result of its review, Staff found that "HL&P is financially qualified to design and construct the Allens Creek Nuclear Generating Station Unit 1". (Staff's witness Petersen, p. 20-8, following Tr. 20433).

23 Cross examination of Messrs. Dean and Petersen revealed no fundamental flaws in the projections of either the Applicant or Staff. The Commission has previously recognized the uncertainties in long-term financial forecasting. See Seabrook, supra, 7 NRC at 19. Undoubtedly those uncertainties exist in this case. However, nothing in the record would lead to a conclusion contrary to that reached by the Staff. The Board finds that Applicant has a reasonable financing plan in light of relevant circumstances, has satisfied the requirements of 10 C.F.R. §50.33(f) and is financially qualified to construct ACNGS.

Bishop Contention 1: Population Projections

24 In its March 10, 1980, Order the Licensing Board admitted Bishop Contention 1 to the extent that it challenges the Applicant's methodology in arriving at its population projections. The contention admitted by the Board asks further whether, in light of new data, these projections need to be reassessed to determine whether the Commission's population criteria will be satisfied. (March 10 Order at 48-49). The new data upon which Mr. Bishop relied in formulating his contention con-

sisted of a population study performed by the Rice Center in 1978. (March 10 Order at 48; Tr. 910-912).

25 In response to the contention, the Applicant presented the testimony of Dr. William T. White, a sociologist and demographic consultant employed by the consulting firm of Dames & Moore. Dames & Moore was retained by HL&P to perform each of its projections accompanying this CP application. (Applicant's witness White, pp. 3-4, following Tr. 8910). The NRC Staff presented the testimony of Messrs. Charles M. Ferrell and Leonard Soffer, both of whom are employed in the Siting Analysis Branch of the Nuclear Regulatory Commission. Mr. Bishop did not attend the hearing sessions to cross-examine the witnesses presented on his contention, and did not present any witnesses of his own.

26 Dr. White reviewed the population projections presented by the Applicant at different times during the licensing of the ACNGS, including projections developed by the Rice Center. (Applicant's witness White, pp. 4-5, following Tr. 8910). The Applicant's first set of population projections were contained in its Environmental Report and were based upon work performed by the Houston-Galveston Area Council (HGAC) in 1972. After reviewing this data, as well as the Staff's population

projections, the Licensing Board found, in its Partial Initial Decision (PID) in 1975, that the Applicant met the NRC's population criteria contained in 10 C.F.R. Part 100. (PID, 2 NRC 776, 798). The population projections in the Applicant's Environmental Report Supplement (Appl. Exh. 13), submitted after the ACNGS application was reactivated, were prepared by Dames & Moore in 1977 and were based upon projections developed by the Texas Water Development Board (TWDB). Later, and in response to the Bishop Contention, Dames & Moore prepared two additional sets of projections, both of which were based upon studies performed by the Rice Center.*/ Each of the four sets of projections were allocated into population wheels and were presented in comparative figures for different time periods and

*/ Two sets of projections were prepared based upon the Rice Center study because it was initially impossible to make a meaningful comparison of the Rice Center projections with those contained in the Environmental Report and Environmental Report Supplement. This was because the Rice Center study did not make projections beyond the year 2000 and because the Rice Center study did not include all of the counties within a 50 mile radius of the Allens Creek site. Therefore, Dames & Moore arranged with Rice Center to use their model to perform an analysis comparable to those performed for the ER and ER Supplement and used the most current data for input into Rice Center's computer so that the results are even more current than the study cited by Mr. Bishop. (Applicant's witness White, pp. 6-7, following Tr. 8910).

distances from the plant site. (Applicant's witness White, Exh. WTW-1, following Tr. 8910). In general, the largest projections were the 1972 projections presented in the Environmental Report, and considered by the Board in writing the PID. The lowest projections, generally, are those based upon the TWDB data and presented in the ER Supplement. (Applicant's witness White, pp. 4-5, following Tr. 8910).

27 To assess the reasonableness of the Rice Center projections, Dames & Moore reviewed the computer model created by Rice Center, and concluded that this model was one of the most advanced models available for developing population projections. (Applicant's witness White, pp. 4-5, following Tr. 8910). For example, the Rice Center model takes into account differences in growth patterns throughout the region, and Dames & Moore assessed the accuracy of these patterns by doing a field investigation around the site. This investigation demonstrated that there is no reason to change any of the methodology in the Rice Center program. Dames & Moore also performed a field study to check the Rice Center allocation model and determined that there were no physical or planning constraints which might affect growth in the area between Allens Creek and Houston

which were not already considered. (Applicant's witness White, pp. 4-6, following Tr. 8910). Dames & Moore also reviewed in detail the economic variables utilized in the Rice Center model, and the results of Dames & Moore's assessment are contained in Appendix A to Exhibit WTW-1, attached to Mr. White's testimony.

28 For the zero to 10 mile annulus, Dames & Moore did not rely upon population projections obtained through demographic or economic analyses of the region. Instead, Dames & Moore, in 1972 and again in 1980, took aerial photographs of the area within ten miles of the Allens Creek site and counted the number of residences within that area to provide a detailed basis for population projections within this annulus. (Applicant's witness White, p. 8, following Tr. 8910). The 1980 photographs show that there has been about a 3,100 person increase within ten miles of the site as compared to 4,760 person increase forecasted in the 1972 projections previously considered by the Board. Based upon this information, Dames & Moore concluded that their projections of growth around the immediate vicinity of the site have been very reasonable. (Id.). On redirect examination, Mr. White testified that aerial photographs are the most accurate means for counting residences

within an area of this size. (Applicant's witness White, Tr. 9146).

29 Dames & Moore compared all four sets of population projections with the population criteria contained in Regulatory Guide 4.7 and with the draft population criteria presented for review in NUREG-0625. These analyses show that the Allens Creek site meets all of the population criteria regardless of which one of the four sets of projections is used. (Applicant's witness White, Exh. WTW-1, pp. 7-11, following Tr. 8910).

30 The NRC Staff testimony discussed applicable NRC population criteria and confirmed that the present projections meet these criteria and will continue to do so over the lifetime of the plant. The criteria discussed by the Staff include those contained in 10 C.F.R. Part 100 and in Regulatory Guide 4.7. (Staff's witnesses Farrell and Soffer, p. 2, following Tr. 17993).*/

---

*/ The Staff did not consider the criteria in NUREG-0625 because, as the testimony states, these are not even proposed siting guidelines. The draft criteria listed in this NUREG merely gives numerical examples to illustrate the types of concepts the Staff is now using in response to the Commission's announced intention to revise 10 C.F.R. Part 100. (Staff's witnesses Ferrell and Soffer, p. 6, following Tr. 17993).

31 In its original SER the NRC Staff found that the population around the Allens Creek site met the requirements of 10 C.F.R. Part 100 with respect to exclusion area, low population zone, and population center distance requirements. (Staff's witnesses Ferrell and Soffer, p. 7, following Tr. 17993). After the Applicant resubmitted population data in 1977, the Staff reported new findings in its SER Supplement 2 issued in March 1979, and re-evaluated the Applicant's population projections. Included in this re-analysis was a review of the "population center distance" requirement to determine whether towns closer to the site than the City of Rosenberg (about twenty miles away) ought to be considered th.. new population center distance. The Staff concluded that based upon new projections, the possibility that either Sealy or Katy (7 and 19 miles from the site, respectively) would become population centers could not be ruled out although it was considered unlikely that Sealey would become the population center during the plant's lifetime.*/ Nevertheless, the Staff

*/ Pursuant to NRC regulation, the population center must have a population of greater than 25,000 people. (10 C.F.R. § 100.11(a)(3)). The present population of Sealy, Texas, is 3,888 persons. (Staff's witnesses Farrell and Soffer, p. 9, following Tr. 17993).

concluded that even if one of these cities became the population center, the distance would still be greater than 1-1/3 times the distance to the outer boundary of the low population zone, and Part 100 requirements would still be met. (Staff's witnesses Ferrell and Soffer, pp. 8-9, following Tr. 17993). The Staff again re-evaluated its conclusion based upon preliminary census data which became available in 1980. This data further demonstrates that the conclusion in the SER Supplement 2 remains unchanged, and the Staff reaffirmed in its testimony that the requirements of Part 100 will be satisfied. (Staff's witnesses Ferrell and Soffer, pp. 9-10, following Tr. 17993).

32 The Staff also examined the Applicant's population data, including its sources and methodology and assessed whether present and projected population densities will exceed the "trip" levels of Regulatory Guide 4.7.*/

---

*/ Regulatory Guide 4.7 provides that population density, including weighted transient population, projected at the time of initial operation of a nuclear power station should not exceed 500 persons per square mile averaged over any radial distance out to thirty miles and that projected density over the lifetime of the facility should not exceed 1000 persons per square mile averaged over any radial distance out to thirty miles. If these trip levels are surpassed, special attention is to be given to the consideration of alternative sites with lower population densities. (U.S. NRC Regulatory Guide 4.7 (Rev. 1, Nov. 1975), "General Site Suitability Criteria for Nuclear Power Station."

First, based upon 1970 census data the Staff, making use of its own computer program employing a copy of the 1970 census tape, independently confirmed that the Applicant's data presented in its ER are reasonable. (Staff's witnesses Ferrell and Soffer, pp. 10-11, following Tr. 17993). While the Applicant's numbers for the zero to 5 mile annulus were approximately 35% lower, the Staff noted that the Applicant's use of a house count as opposed to a computer program renders Applicant's numbers more reliable. (Staff's witnesses Ferrell and Soffer, p. 12, following Tr. 17993). The Staff also compared the latest population projections of the Applicant utilizing the Rice Center model with preliminary population data obtained from the 1980 census. The Staff found that Applicant's and Staff's data from zero to 30 miles were in reasonably close agreement and that both sets of projections showed densities well below the trip levels of Regulatory Guide 4.7. (Staff's witnesses Ferrell and Soffer, pp. 13-14, following Tr. 17993).

33 The Staff also utilized the Applicant's most recent population projections to determine whether Reg. Guide 4.7 "trip levels" would be met, assuming 1990 to be the estimated beginning of plant life and 2030 to be the end of plant life. Extrapolating Applicant's projections

from the year 2020 to the year 2030 (assuming the same growth rate) the Staff found that projected population density will be well below the trip levels of 500 persons per square mile in 1990 and 1,000 persons per square mile over the life of the plant and that Regulatory Guide 4.7 levels will not be exceeded. (Staff's witnesses Ferrell and Soffer, pp. 17-18, following Tr. 17993).

34 The Staff assessed the reasonableness of the data sources utilized by Applicant. The Staff noted that the sources used by the Applicant are all governmental groups or private institutions which are independent of it and that these groups are typically interested in examining future population growth for a variety of reasons. Moreover, the Applicant's projections make use of data related to economic activity, observed growth patterns, transportation networks, and other appropriate information about the ACNGS region and incorporate these data by means of a suitable methodology. (Staff's witnesses Ferrell and Soffer, p. 15, following Tr. 17993). Moreover, the Staff compared Applicant's data with those obtained from other independent sources. In the Staff's SER dated November 1974, it compared Applicant's projections with independent projections made by the U.S. Department of Commerce, Bureau of

Economic Analysis for a 17 county area including the Houston-Galveston area and surrounding counties. The Staff therein found the Applicant's projections to be in reasonable agreement with those of the Bureau of Economic Analysis. (Staff's witnesses Ferrell and Soffer, p. 15, following Tr. 17993; Staff Exh. 1, pp. 2-8). Since this original comparison, the Staff has also obtained more recent projections by the Texas Department of Water Resources (TDWR) which were published in January, 1980, and unpublished projections out to the year 2020 for a number of counties of interest were also obtained from the TDWR. The Staff found that the Applicant's most recent projections are also in reasonable agreement with these most recent projections made by this independent source. (Staff's witnesses Ferrell and Soffer, pp. 15-16, following Tr. 17993). Based upon all of the above, the Staff concluded that the Applicant's projections utilize appropriate data and methodologies and that there is reasonable agreement between Applicant's and Staff's independent projections. (Staff's witnesses Ferrell and Soffer, p. 17, following Tr. 17993).

35 Based upon the evidence discussed above, the Board finds that the methodology and data employed by the

Applicant, and independently reviewed by the Staff, are reasonable. Rather than providing support for Mr. Bishop's contention, the work done by Rice Center shows that the projections utilized in the 1975 PID were generally conservative. The Board again finds that the ACNGS site complies with the Part 100 criteria, and it finds further that population densities around the site do not exceed the "trip levels" of Regulatory Guide 4.7.

Bishop Contentions 4, 5, 7, 9

36 Intervenor Bishop has raised several contentions regarding the potential rupture of the Texas Electric Service Company's (TESCo) 24-inch natural gas pipeline which is to be relocated during construction of ACNGS. Mr. Bishop alleges in his Contentions 4 and 5 that the Applicant has not adequately analyzed the risk to the population of the town of Simonton and the Valley Lodge subdivision which could occur as a result of the rupture of the 24-inch natural gas pipeline. Intervenor further contends that due to the instability of the Brazos River bank the probability of the rupture of the relocated 24-inch gas pipeline has been increased. In Contention 7, Intervenor alleges that a rupture of the

relocated 24-inch natural gas pipeline could cause a breach of the cooling lake dam resulting in the release of water from the ACNGS cooling lake. In Contention 9, Intervenor alleges that Applicant has underestimated the risk to the plant from a postulated detonation of the gas cloud due to the rupture of the 24-inch natural gas pipeline.*/

37 The 24-inch natural gas pipeline which is the subject of Bishop Contentions 4, 5, 7 and 9, will be relocated in order to avoid the ACNGS cooling lake. The relocated pipeline will pass approximately 700 feet east of the cooling lake dam between the dam and the Brazos River. The relocated pipeline will pass approximately 9300 feet northeast of the nearest Category I structure at ACNGS. (Applicant's witness Iotti, p. 3 following Tr. 11407). The Applicant has determined from information supplied by TESCo that the pipeline will carry only natural gas which is at least 90% methane (see Exhibit RCI-2 attached to the testimony of Applicant's witness Iotti, following Tr. 11407). There is no indication that materials other than natural gas will be carried in this pipeline. (Applicant's witness Iotti at 3-4, following Tr. 11407).

---

*/ Intervenor Bishop presented no evidence on these or any other of his contentions.

There are no other materials normally carried in this type of pipeline which would present a greater potential hazard to the plant than methane. (Applicant's witness Iotti, Tr. 11599-601; 11619).

38 The Board finds that the 24-inch natural gas pipeline as proposed to be relocated will not present a hazard to either the plant, the cooling lake dam, or the communities of Simonton and Valley Lodge. First of all, although the analysis takes no credit for it, steps have been taken to insure that a leakage or rupture of this pipeline will not occur. TESCo will provide the industry standard means of cathodic protection for the pipeline along the segment that will be relocated. In addition, TESCo will install pressure sensors which will signal a rupture or leak in the pipeline, and is required to install an isolation valve which will prevent the escape of significant quantities of natural gas. (Applicant's witness Iotti at 4-5, following Tr. 11407).

39 With respect to the plant structures, the Applicant has analyzed, using extremely conservative assumptions, a postulated rupture of the natural gas pipeline. This analysis is set forth in Appendix 2.2-A of the ACNGS PSAR. (App. Exh. 27). The analysis has assumed that a

detonable cloud of methane (ignoring buoyancy) could extend 4200 feet from the assumed break in the pipeline. The overpressures resulting from a postulated detonation of such a cloud can be withstood by the safety-related plant structures. (Applicant's witness Iotti at 5-8, following Tr. 11407; Tr. 11622). In addition, the Applicant analyzed whether missiles would pose any hazard to the plant if a methane cloud detonated, and found that no such hazard would occur. (Applicant's witness Iotti at 8-9, following Tr. 11407; Tr. 11519). It should be noted, however, that the testimony indicated that it would be virtually impossible for a methane cloud to detonate because of the lack of high energy ignition sources in the vicinity of the pipeline. (Staff's witness Campe at 6, following Tr. 11647; Tr. 11662-65). Moreover, there have been no instances of open air cloud detonation from methane air mixtures. (Applicant's witness Iotti, Tr. 11626).

40 With respect to the potential hazards to the ACNGS cooling lake dam, the Applicant analyzed the effect on the dam of a postulated rupture of the 24-inch natural gas pipeline. This analysis, again using extremely conservative assumptions, resulted in a maximum peak overpressure of 2.1 psi, which the cooling lake dam is

designed to withstand. (Applicant's witness Iotti at 11-12, following Tr. 11407; Tr. 11531-32; 11533-34). Further, the Applicant analyzed whether a postulated missile could cause any damage to the ACNGS dam as a result of a detonation of a natural gas cloud, and concluded that there was no risk of damage to the dam from postulated missiles. (Applicant's witness Iotti at 13, following Tr. 11407). Even in the unlikely event that the dam were breached by a detonation of a methane cloud shut-down cooling would be provided by water stored in the seismic Category I portion of the Ultimate Heat Sink, and there would be only a minimal effect offsite from potential flooding. (Staff's witness Campe at 6, following Tr. 11647).

41 With respect to Intervenor's allegations that the Valley Lodge subdivision and the community of Simonton are at risk as a result of the relocated natural gas pipeline, the Applicant's witness testified that using the same conservative assumptions as he did with respect to the potential impact of the detonation on plant structures, the pipeline posed no hazard to these communities. (Applicant's witness Iotti at 14-15, following Tr. 11407). This analysis demonstrates that a maximum overpressure of 0.4 psi could result from

detonation of a cloud of methane gas coming from a rupture in the pipeline. This overpressure would result in no damage to the communities of Valley Lodge or Simonton beyond possible window breakage. (Applicant's witness Iotti, pp. 14-15, following Tr. 11407; Tr. 11628). Moreover, there is little likelihood that such a gas cloud would detonate because of the lack of a sufficient ignition source in the vicinity of the pipeline. (Staff witness Campe at 4, following Tr. 11647). In addition, there is no danger to Valley Lodge or Simonton because of possible asphyxiation from a postulated release of natural gas from the pipeline. Concentrations of natural gas from such a postulated release will be far below that which is needed for asphyxiation and also far below the concentrations given as a threshold for discomfort by the time the gas could reach Valley Lodge or Simonton. (Applicant's witness Iotti at 15-16, following Tr. 11407; Staff's witness Campe at 2-3, following Tr. 11647).

42 Finally, there will be no hazard to these communities from a potential deflagration of a postulated methane cloud because the maximum distance at which this deflagration would cause detrimental effects to the people in these communities is significantly less than

the distance to these communities. (Staff's witness Campe at 3, following Tr. 11647).

43 With respect to Bishop Contentions 4 and 5 which allege that the Brazos River bank may erode causing the 24-inch natural gas pipeline to rupture, the Applicant will stabilize the Brazos River bank in the area where the 24-inch gas pipeline will be relocated. That relocation is between the Allens Creek cooling lake dam and the Brazos River. (Applicant's witness Mercurio, pp. 17-20, following Tr. 11407; App. Exh. 20). The stabilization program will be undertaken either by Applicant itself or in conjunction with the State of Texas. The State of Texas is concerned about maintaining Farm to Market Road 1458 which lies between the relocated 24-inch natural gas pipeline and the Brazos River bank. The Board finds that since the river will be stabilized, there is no risk that the 24-inch natural gas pipeline will rupture due to erosion of the Brazos River bank as alleged by Mr. Bishop. It should be noted, however, that both the Applicant and Staff analyses assume that there is a rupture of the pipeline. (E.g., Staff's witness Campe, p. 5, following Tr. 11647).

Bishop Contention 6/Board Question 12: LPG Pipeline

44 Bishop Contention 6 alleges that the Shell Oil Company's 6-inch liquid petroleum gas (LPG) pipeline, which is located approximately 8000 ft. from the nearest Category I structure, could rupture and release an explosive gas cloud which would travel close to the plant, explode and thereby damage the plant.

45 Conservative analyses performed by the Applicant and the Staff demonstrate that safety related structures at ACNGS can withstand the overpressures which could be generated by a postulated detonation of such an LPG cloud. In analyzing the potential effects on ACNGS from a postulated detonation of an LPG cloud, the operation of Shell's leak detection system and subsequent operation of the pipeline isolation valves were essentially ignored even though the pipeline can be isolated and the pumps stopped within approximately 8 minutes. (Applicant's witness Iotti, p. 5, following Tr. 17135; Tr. 17234-35; App. Exh. 26). Instead, a conservative assumption was made that 100 lbs./sec. of propane flows continuously from the break. (Applicant's witness Iotti, p. 6, following Tr. 17135; Tr. 17234-35). Under the worst case conditions, it was assumed that

the LPG would escape from the ruptured pipeline, that a propane gas cloud would be formed which would move by gravity flow along the Allens Creek depression past the plant to the cooling lake without being atmospherically dispersed. This would result in a propane cloud of approximately 4.3 x $10^6$ ft.$^3$. (Staff's witness Campe, pp. 4-5, following Tr. 17238). Further, it was assumed that the propane cloud could detonate within 1610 ft. from the plant. (Staff's witness Campe at 4, following Tr. 17238).

46 Applicant's analysis shows a peak incident overpressure from detonation of the cloud to be 1 psi to the nearest ACNGS safety related structure (Applicant's witness Iotti at 8, following 17135). Even if reflected overpressures are taken into account (this increases the overpressure by approximately a factor of 2 and takes into account the blast wave being reflected off a structure) the overpressure resulting is about 2.2 psi which is within the 2.3 psi design basis tornado loadings for the ACNGS plant. (Staff's witness Campe at 5, following Tr. 17238; Tr. 17144). The 2.3 psi design basis tornado loadings represent the pressure loadings that the ACNGS safety related structures are designed to withstand, and the structures can probably withstand

much higher loadings (Staff's witness Campe, Tr. 17255-58).

47 The postulated propane gas cloud will also pose no danger to ACNGS structures if it should deflagrate rather than detonate. The closest deflagrable cloud would be 1580 ft. away from the plant and would result in overpressures less than those from a detonable cloud. (Staff witness Campe at 4-5, following Tr. 17238; Tr. 17225). Furthermore, thermal exposure from a deflagration will not have any significant effect on ACNGS structures. (Staff witness Campe at 5, following Tr. 17238). Finally, there is no danger to plant structures resulting from potential missiles following a postulated detonation of a propane gas cloud. If all the mass from generated missiles is assumed to be concentrated in one missile, which then impacts upon the plant, while traveling at its maximum air particle velocity, no damage to plant structures will occur. (Applicant's witness Iotti at 8-9, following Tr. 17135; Tr. 17149-50; Tr. 17213-17).

48 In relation to Board Question 12, which asked whether potentially more dangerous materials might be carried in the pipeline, the testimony reflects that the choice of a propane gas cloud for the analyses represents a conservative choice and represents the greatest potential

hazard to the plant. (Applicant's witness Iotti at 10-12, following Tr. 17135; Tr. 17228-29; Tr. 17280-81). In fact, the Staff assumed in its analysis the TNT equivalency of methane (240%) rather than propane (228%) as an added conservatism even though there is no indication that methane will be carried in the pipeline and in any event, methane would quickly disperse since it is lighter than air. (Staff's witness Campe, p. 6, following Tr. 17238; Tr. 17267; Tr. 17277).

49 The Board is satisfied that a possible rupture of the 6-inch Shell Oil Co. LPG pipeline will not present any hazard to the ACNGS Plant. The Board finds that the analyses performed by both Applicant and Staff are in agreement (Applicant's witness Iotti, Tr. 17136-45) and are based upon conservative assumptions, such as the postulated size of the cloud (Applicant's witness Iotti, Tr. 17202-03) and meterological conditions which would move the cloud toward the plant prior to an assumed detonation. (Applicant's witness Iotti at 5-6, following Tr. 17135; Staff witness Campe at 4, following Tr. 17238). Even if a worst case detonation occurred, the resulting overpressures and thermal effects will be within the design basis of the structures. Accordingly, the Board finds that there is no merit to the allegations set forth in Bishop Contention 6.

Bishop Contention 10: Rupture of Upstream Pipeline

50 Intervenor Bishop alleges in Contention 10 that pipelines crossing the Brazos River upstream from the ACNGS plant could rupture, releasing flammable or corrosive materials into the Brazos River which could then enter the cooling lake and create a hazard to the plant.

51 There is very little risk to plant structures which could result from a postulated rupture of any of the pipelines upstream of the ACNGS plant. (Staff's witness Campe at 9, following Tr. 11647). These pipelines carry either crude oil, natural gas or unprocessed natural gas liquid. (Applicant's witness Carnes p. 20, following Tr. 11407). First of all, makeup water is not taken continuously from the Brazos River. When it is taken, it is first directed to the settling basin where anything that has a specific gravity greater than one will settle out (Applicant's witness Carnes, Tr. 11582). HL&P will also periodically monitor the makeup water and if any contamination of the river is reported, pumping can be stopped. (Applicant's witness Carnes at 21, following Tr. 11407; Staff's witness Campe at 9-10, following Tr. 11647; Tr. 11675). Secondly, the makeup water at the Lake Makeup Structure is taken from the

bottom of the Brazos River, and therefore any substances which are floating on top of the river will not be taken into the intake structure. Crude oil released from upstream pipelines is lighter than water, would tend to float on top of the Brazos River, and would not be taken into the intake structure. (Applicant's witness Carnes at 21-22, following Tr. 11407; Staff's witness Campe at 9, following Tr. 11647). Liquids which are capable of mixing with water would be substantially diluted. (Staff's witness Campe at 9, following Tr. 11647). Even if such contaminants (including emulsions) could enter into the cooling lake from the settling basin, there would not be a significant risk to the plant because the contaminants would be greatly diluted. Moreover, any component degradation or fouling that did result from contact with possible contaminants would be detected by the Applicant during normal plant operation. (Applicant's witness Carnes at 22-23, following Tr. 11407; Staff's witness Campe at 10, following Tr. 11647; Tr. 11582-83; 11633-34; 11692). Accordingly, the Board finds that a rupture of pipelines which cross the Brazos River upstream of the ACNGS Plant will pose no hazard to the ACNGS.

Bishop Contention 17: TNT Detonation

52 In Contention 17, Intervenor Bishop alleges that Applicant has underestimated the effect on ACNGS structures of a rupture and/or detonation of train carloads of TNT.

53 In Section 2.2.3.3.1 of the PSAR (App. Exh. 27), Applicant analyzed the postulated detonation of a train carload of TNT, assuming a conservative load of 200,000 lbs. of TNT. It was determined that the maximum load on Category I structures as a result of this postulated detonation was around 0.55 psi, a value well below the design basis tornado loading of 2.3 psi for Category I structures. (Applicant's witness Iotti, p. 3, following Tr. 17286).

54 Regulatory Guide 1.91 "Evaluations of Explosions Postulated to Occur on Transportation Routes near Nuclear Power Plants," provides that if the peak positive incident overpressure from detonation of a train carload of TNT does not exceed 1 psi, then plant structures will be adequately protected from this event. The Regulatory Guide provides a formula for determining whether the plant is at a "safe" distance from the postulated detonation of the TNT and assumes that a

single railroad box car will detonate 132,000 lbs. of TNT. According to the formula, a plant structure would have to be closer than 2,291 ft. to the assumed point of detonation for damage to result. In the case of ACNGS, the nearest Category I plant structure is 4,230 ft. from the Atchison, Topeka and Santa Fe (AT&SF) Railroad. (Applicant witness Iotti, p. 2, following Tr. 17286; Staff's witness Campe, p. 3, following Tr. 17314). Accordingly, the ACNGS plant is sufficiently far away from the railroad so that the detonation of a car load of TNT would not present any hazard to the proposed plant. (Applicant's witness Iotti, pp. 2-3, following Tr. 17286). In fact, at this distance it would take the simultaneous detonation of up to 10 train carloads of TNT, each loaded with 132,000 lbs. of TNT, to yield a peak incident overpressure of approximately 1 psi. (Staff's witness Campe, p. 3, following Tr. 17314). Even under Applicant's conservative assumption that each train car carries 200,000 lbs. of TNT, it would take the simultaneous detonation of approximately six cars in order to reach 1 psi incident overpressure. (Applicant's witness Iotti, p. 4, following Tr. 17286). Both Staff and Applicant witnesses emphasized, however, that it is extremely unlikely that this number of carloads would be on the same train shipment and would also detonate simultaneously. (Applicant's witness Iotti, p. 4-5, following Tr. 17286; Staff's witness Campe, p. 3, following Tr. 17314; Tr. 17326-27).

55 While TNT was used as the basis for determining the resulting overpressures, Applicant did consider the explosives having yields greater than TNT, but determined that in each case, overpressures were well within the acceptance criteria of 1 psi contained in Regulatory Guide 1.91. (Applicant's witness Iotti, p. 3, following Tr. 17286; Tr. 17291).

56 The Board finds that the Applicant and Staff have not underestimated the potential effects on the plant from a postulated detonation of TNT carried on the AT&SF Railroad, and that the ACNGS structures will withstand such explosions.

Doherty Contention No. 3: Fuel Enthalpy

57 Intervenor Doherty alleges in Contention 3 that the safety design limit of thermal energy for fuel rods is too high. The allegation is based on the results of tests on two General Electric fuel rods which had shown failures during testing. The contention was examined

at considerable length during the hearing. Applicant's witnesses, Mr. Kevin Holtzclaw and Dr. Richard Williams, testified for a full day and part of an evening session. (Tr. 11718-12035). Staff's witness, Dr. Ralph Meyer, testified for parts of two days (Tr. 13995-14083; 14093-14240). Nearly another day of hearing time was taken up by Mr. Doherty's attempt to have TexPirg's attorney, Mr. Scott, testify on this contention. (Tr. 14244-14286; 14309-14399). Much of this time was consumed in an effort to clear-up Mr. Doherty's confusion between the concepts of safety design limit and fuel failure threshold.*/ (E.g., Tr. 11922-11945; 14061-14083; 14149-53).

58 The NRC's safety design limit of 280 calories per gram ("cal/gm") corresponds to a value which sets an upper limit on the permissible amount of heat deposited in a fuel rod during a reactivity insertion accident ("RIA").

---

*/ After hearing cross-examination on Mr. Scott's direct testimony, Mr. Doherty admitted that Mr. Scott had defined the term "safety design limit" so as to equate it with the term "fuel failure threshold", which Mr. Doherty finally admitted was inconsistent with the meaning used in the contention. Accordingly, the Board concluded that Mr. Scott's testimony should be stricken in light of the irreconcilable conflict between the testimony and Mr. Doherty's contention (Tr. 14394-99). Mr. Doherty made no effort to present an affirmative case on his other contentions.

This limit assures that large pressure pulses will be avoided and that coolability of the fuel will be maintained. The fuel failure threshold of 170 cal/gram is not a limit at all. It is simply a value that is used for radiological dose calculations. (Staff's witness Meyer, Tr. 14020-21). Tests performed to establish the fuel failure threshold have no relationship to the establishment of the safety design limit. This is the case with the tests cited in Doherty Contention No. 3. (Staff's witness Meyer, Tr. 14022-23; Applicant's witness Holtzclaw, Tr. 11922-23; 11943-45; 12201-7; 12222-25). Indeed, none of the consequences described in the contention resulted from these tests. (Staff's witness Meyer, Tr. 14023).

59 Recent tests and re-examination of the SPERT tests conducted by EG&G Idaho, Inc. at the Idaho National Engineering Laboratory suggest that a radial average peak fuel enthalpy limit of 230 cal/gm might be more appropriate for ensuring that fuel rods maintain a coolable geometry during a reactivity insertion accident. (Applicant's witness Holtzclaw, at p. 7, following Tr. 11750; Tr. 11945; 12208-12; Staff's witness

Meyer, at p. 3, following Tr. 14019; 14057-59).*/ The NRC's safety design limit of 280 cal/gram is given as a radially averaged fuel enthalpy; however, the limit was derived from the SPERT tests which were reported as total energy insertions. The SPERT 280 cal/gm value equates to 230 cal/gm on a radially averaged basis. Nonetheless, Dr. Meyer, who is the Section Leader of the NRC's Reactor Fuels Section, testified that it is not necessary to change the NRC's safety design limit because a recent study by the Brookhaven National Laboratory shows that actual enthalpy values are in the range of 50 to 100 cal/gm. (Staff's witness Meyer, p. 3, following Tr. 14019).

60 This conclusion is affirmatively supported by Mr. Doherty's own evidence. Doherty Exhibits 3 and 3A are papers prepared under the principal direction of Mr. P. E. MacDonald, who is the Manager of the LWR Fuel Research Division in the Thermal Fuels Behavior Program at EG&G Idaho, Inc. These papers conclude that the NRC should consider adopting a safety design limit of 240 cal/gm radial average peak fuel enthalpy. However, Mr. MacDonald concluded:

---

*/ However, Dr. Meyer testified that he still did not believe that unacceptable consequences would occur at the level of 280 cal/gram radial average peak fuel enthalpy. (Tr. 14027-32).

"Light water reactor control systems are presently designed such that if a reactivity initiated accident does occur, the resulting peak fuel enthalpy will be below 110 cal/gram. The PBF results indicate that there is no safety problem with respect to loss-of-coolable geometry, fuel failure propagation, or molten fuel coolant interaction as a result of RIA in a commercial power plant."

(Doherty Exhibit 3A, p. 5; see also Doherty Exhibit 3, pp. 601-2; Tr. 14232-33).

61 This is consistent with the results reached by GE for the ACNGS fuel design. The most severe RIA is a rod drop accident, with a calculated maximum total energy deposition of 135 cal/gm.*/ (Applicant's witness Holtzclaw pp. 3-4, following Tr. 11750). This is a very conservative analysis that does not account for moderator feedback. (Applicant's witness Holtzclaw, p. 5, following Tr. 11750). If the number is converted to peak radial enthalpy and adjusted to account for moderator feedback and heat transfer the result is approximately 60 cal/gm. (Tr. 11765-66).**/

---

*/ As discussed in the findings on Doherty Contention 24, it is extremely unlikely that there ever will be a rod drop accident at ACNGS.

**/ GE's model is an adiabiatic model which does not account for the heat transfer that actually takes place. It is, therefore, a very conservative model (Tr. 11774-85; 14110; 12208-11).

62 In sum, the Board finds that the peak radial average fuel enthalpy for a rod drop accident is far below the present safety design limit (280 cal/gm) or the possible revision to that limit that has been suggested by researchers at EG&G (240 cal/gm). Accordingly, any question as to whether the NRC's safety design limit should be in the range of 230 to 280 cal/gm is irrelevant to the question of whether ACNGS can be operated safely. The actual values for the ACNGS rod drop accident are far below any revision to the safety limit that would arguably result from the EG&G tests.

Doherty 5: Suppression Pool Uplift

63 Intervenor Doherty alleges, in Contention 5, that the control rod drive mechanism hydraulic control units (HCU) and the transversing incore probe (TIP) may be damaged because of the forces of a vertical water swell in the suppression pool following a loss-of-coolant accident (LOCA).*/

*/ Applicant filed a motion for summary disposition on this issue which was denied by the Board in its Second Order Ruling Upon Motions For Summary Disposition, dated September 1, 1981, for the reasons stated on pages 30-32.

64 The "poolswell" phenomenon can be caused when, following a postulated LOCA, steam and air from the drywell is directed through the horizontal vents into the suppression pool water. This mixture of steam and air causes a bubble to rise through the suppression pool water, break through the water surface, and to form a froth of water prior to the time it falls back into the suppression pool. (Applicant's witness Stancavage, pp. 2-4 of Attachment PPS-1, following Tr. 20297; Tr. 20312). Poolswell occurs in two stages ("bulk" swell and "froth" swell) and will cause two different types of loads to be placed upon structures and components located above the suppression pool water: impact loads and drag loads. With respect to bulk poolswell and impact loads, test data show that */ after the pool has risen approximately 12 ft., the slug thickness has decreased to 2 feet or less and the impact loads are significantly reduced and will ultimately terminate at a level below

*/ The General Electric Mark III containment pressure suppression testing program was initiated in 1976 with a series of small-scale tests and culminated in full-scale testing at the Mark III Pressure Suppression Test Facility (PSTF). General Electric used the test data to develop hydrodynamic loading conditions in the GE Mark III reference plant pressure suppression containment system during the postulated LOCA. (Stancavage, pp. 4-7).

18 ft. above the pool. (Applicant's witness Stancavage, p. 7 of Attachment PPS-1, following Tr. 20297). Between 18 and 19 ft., there will be a transition from water to froth and the froth impingement load is conservatively assumed to be 15 psig. (Id., p. 8; Staff's witness Fields, Tr. 19320). Accordingly, General Electric has determined that the HCUs should be placed higher than 19 ft. above the normal suppression pool surface. (Applicant's witness Stancavage, pp. 7-8 of Attachment PPS-1, following Tr. 20297). Above 19 ft. the HCUs will experience only an impulsive loading followed by a pressure differential loading. This is due to the momentum of the suppression pool froth and the transient pressure in the space between the pool surface and the HCU floor. (Applicant's witness Stancavage, pp. 8-9 of Attachment PPS-1, following Tr. 20297).

65 The HCU modules are located on a steel platform which is 22 ft. 5 in. above the normal suppression pool surface. This steel platform is supported by beams and girders which span the annulus between the drywell and the containment; the bottom of the floor girders are approximately 20 ft. above the surface of the suppression pool. (Applicant's witness Nuta, p. 2, following

Tr. 20244). The floor girders will only be subjected to the impact of froth impingement resulting from the LOCA loads which are discussed above. (Applicant's witness Nuta, p. 2, following Tr. 20244; Tr. 20261). The Applicant has analyzed the steel platform which supports the HCU Modules to determine the effect of the froth impingement due to LOCA loads.*/ This analysis demonstrates that the HCU floor response spectra peaks will be significantly lower than the dynamic capability of the HCU modules. (Applicant's witness Nuta, p. 3, following Tr. 20244). The HCU modules themselves are designed to withstand dynamic loads in excess of 15g vertical and 5.9 to 11.9g horizontally, and therefore will not be damaged by the poolswell loads resulting from a LOCA. (Applicant's witness Stancavage, p. 1, following Tr. 20297).

66 Intervenor Doherty also alleges that pool swell will cause damage to the TIP. The TIP station is located at an elevation of about 6 ft. above the normal suppression pool surface. The station consists of a concrete cantilever platform which is designed so that the

---

*/ Vibratory response of the HCU floor to the froth impingement would subsequently transmit a load to the HCU modules. (Applicant's witness Stancavage, pp. 9-10, following Tr. 20297).

bottom surface of the cantilever is immersed in the suppression pool. In addition, it is sloped upward and functions as a deflector for the pool swell impact. (Applicant's witness Nuta, p. 3, following Tr. 20244; Tr. 20270-71).

67 The TIP station is designed to experience a maximum drag load of 11 psid and a 21.8 psid bubble pressure load. Moreover, the TIP is protected by the cantilever structures which are specifically designed to absorb this particular loading. (Applicant's witness Stancavage, p. 10 of Attachment PPS-1, following Tr. 20297). It should be noted that the TIP does not perform a safety related function, but is designed to calibrate the local power range monitors. (Applicant's witness Stancavage, pp. 10-11 of Attachment PPS-1, following Tr. 20297). Accordingly, the Applicant concludes that although the TIP does not perform any safety function following a LOCA, it has been designed to absorb and be protected against this particular accident loading. (Id.)

68 The Staff concluded in the SER for GESSAR that poolswell loads for Mark III containments are sufficiently known and defined for purposes of issuance of a construction permit. While most of the Mark III design specifications

for LOCA-related suppression pool loads have been accepted by the Staff, the Staff is continuing its review of the final loads (Task Action B-10) and expects to issue a NUREG document on these loads in February 1982. However, the Staff believes that if any changes to the design load criteria are required as a result of this review, which is nearly complete, they will be minor and easily accommodated by plants which are under construction. (Staff's witness Fields, p. 3, following Tr. 19298; Tr. 19314-15).

69 Finally, in response to the Board's concern stated on page 32 of its Second Order Ruling Upon Motions for Summary Disposition, both Applicant and Staff witnesses stated that there are no mechanisms whereby one or more safety relief valves can actuate following the onset of a LOCA. Therefore, simultaneous SRV operation with a LOCA-caused poolswell need not be considered. (Staff's witness Fields, p. 3, following Tr. 19298; Applicant's witness Hucik, pp. 2-3, following Tr. 20297).

70 The Board finds that, contrary to Mr. Doherty's allegation, the evidence shows that neither the HCU's nor the TIP will be damaged due to poolswell following a LOCA. If changes in the poolswell load definitions result after completion of the Staff's review, these changes

can be accommodated in the ACNGS design prior to the operating license stage.

Doherty Contention 6: Recirculation Pump Overspeed

71 In Contention 6, Intervenor Doherty alleges that an adequate basis has not been provided to assure that an overspeed of the recirculation pump causing destructive missiles will not occur. The contention states that the Applicant has committed to provide a decoupler to prevent overspeed of the pump motor, but the potential for pump impeller missiles exist.

72 The potential for overspeed of the recirculation pump exists if there is a break in the piping of the recirculation loop. This is because the reactor pressure is higher than the drywell pressure, and accordingly, if there is a break in the recirculation loop piping, reactor water would flow from the vessel through the break in the piping. Depending on where the break is located, reactor water could flow through the recirculation pumps either in the forward direction if the break were located in the discharge line, or in reverse direction if the break were located on the suction line. In both cases, the pump will act as a hydraulic

turbine and will be subject to overspeed. Because of the hydraulic characteristic of the impeller design in the recirculation pump, an overspeed in the reverse direction would exert greater forces on the impeller than flow in the normal direction. If the overspeed conditions cause stresses greater than the tensile strength of the impeller material, the impeller would fail causing the missiles of concern in this contention. (Applicant's witness Dillman, pp. 3-4, following Tr. 14535).

73 The original design of ACNGS included a decoupler which would decouple the recirculation pump motor from the pump impeller. The design concept was to protect the pump motor from overspeed conditions (Staff's witness Leung, Tr. 13052). The present ACNGS design now calls for the decoupler to be deleted from the design. The reason for this deletion is that the decoupler is no longer necessary due to the low probability of missile generation and the minimal consequences of such generation. In addition, the decoupler adds undesirable complexity and service requirements even if it should serve its intended purpose. (Applicant's witness Dillman, p. 5, following Tr. 14535). Accordingly, Applicant provided an analysis of both recirculation pump motor overspeed and recirculation pump overspeed

and the missiles thereby created.*/

74 The Applicant's conservative analysis shows that missiles generated by failure of the motor rotor will impact upon the stator and the stator frame and will not be ejected from the motor. In the case of the pump impeller, the analysis demonstrates that the missiles will not penetrate the pump casing, although there is a possibility impeller missiles could escape from the broken end of the pipe. (Applicant's witness Dillman, pp. 5-6, following Tr. 14535). However, even if a recirculation pump impeller missile could be ejected through the recirculation loop piping, the analysis further demonstrates that no unacceptable damage would occur. (Applicant's witness Dillman, p. 7, following Tr. 14535).

---

*/ The Staff has not completed its review of whether the decoupler is necessary, and has indicated that it will do so when it completes its review of generic issue B-68, Pump Overspeed During a LOCA. (Staff witness Leung, pp. 3-4, following Tr. 13038). However, the Staff believes that recirculation pump overspeed with generation of missiles is a low probability event, and consequently, has not put a high priority on the resolution of this issue. In fact, plants are currently being licensed to operate without the decoupler pending resolution of B-68. With respect to ACNGS, the Staff believes that sufficient basis exists to issue a CP, notwithstanding the resolution of B-68, because the Applicant has committed to adopt the generic resolution. (Id, pp. 3-4).

75 Based on the foregoing uncontroverted evidence, it is clear that the issue of the effects of pump overspeed during a LOCA is still an unresolved generic issue. Furthermore, resolution of this generic issue has been given a low priority by the Staff based, in part, on its safety and environmental significance. The insignificance of the safety consequences of recirculation pump overspeed has been demonstrated by Applicant's witness from GE and concurred with by the Staff. (Dillman at 6-9; Leung at 3). This demonstration shows that it is an extremely unlikely event that pump overspeed will occur. In fact, an overspeed event with missile generation has never occurred (Tr. 13042, 13045, 13049-50). If pump overspeed does occur, conservative analyses show that motor missiles will not penetrate the motor housing, and impeller missiles will not penetrate the pump casing except through the pipe break opening. If missiles should eject into the drywell, the postulated projections will not cause any damage to safety systems. Accordingly, the Board concludes that the probability of damage to safety releated equipment from overspeed of both the motor and the impeller of the recirculation pump without a decoupler is acceptably low to permit construction of ACNGS.

Doherty Contention 7: LPCI Cold Slug

76 In Contention 7, Intervenor Doherty claims that injection of water from the suppression pool following an accident will result in an increase in core reactivity. The water would be injected into the core through the low pressure coolant injection (LPCI) system. Mr. Doherty's basic concern is that when the LPCI is operating, the water from the suppression pool injected into the core will be colder than the water in the core, thereby causing an increase in reactivity which he believes could lead to a possible fuel melt.

77 Both Applicant and Staff testified that reactivity inserted due to injection of suppression pool water via the LPCI will not exceed the shutdown margin, *i.e.*, the margin required to keep the reactor subcritical after shutdown. (Applicant's witness Eckert, pp. 5-9, following Tr. 13901; Staff's witness Brooks, pp. 2-3, following Tr. 14442).

78 The Applicant has analyzed injection of suppression pool water following a loss-of-coolant accident (LOCA), Reactivity Insertion Accident (RIA), and Anticipated Transient Without Scram (ATWS), and has determined that for these three events, reactivity insertion due to

injection of LPCI cold water will not exceed the shutdown margin of the reactor. Following a LOCA, the control rods are inserted into the reactor before ECCS injection begins; the analysis shows that the addition of cold ECCS water will not insert sufficient reactivity to exceed the shutdown margin and the core will remain subcritical. (Applicant's witness Eckert, pp. 6-7, following Tr. 13901; Tr. 13913; Staff's witness Brooks at 2, following Tr. 14442). In fact, the shutdown margin will be maintained no matter how cold the water is when it is injected into the core after a scram. (Staff's witness Brooks, p. 3, following Tr. 14442; Tr. 14443). For the RIA, there will be no ECCS operation since there is no loss of water from the reactor itself. (Applicant's witness Eckert, p. 7, following Tr. 13901). For the ATWS event, the reactor is shutdown by tripping the recirculation pumps in combination with either injection of boron into the reactor vessel or by an alternative method of control rod insertion. Only in the case of boron injection will the ECCS operate before the reactor is brought to a subcritical state. In this case only the High Pressure Core Spray (HPCS), and not the LPCI, will be utilized because reactor pressure will be above the pressure required for LPCI injection. Applicant has determined, based on analysis, that reactivity increase due to the HPCS injection, would be neglibible. (Applicant's witness Eckert, pp. 7-8, following Tr. 13901).

79 Finally, the Applicant has analyzed an inadvertant initiation of the ECCS during normal low power operation. This analysis demonstrates that after injection of ECCS water into the core, there will be an increase in core power, but when this power level reaches the intermediate range monitor scram set point, a scram will be initiated which will terminate the power increase and thus prevent any damage to the fuel. (Applicant's witness Eckert, p. 8, following Tr. 13901).

80 The Board finds that there will be no fuel damage as a result of injection of suppression pool water into the core by the LPCI.

Doherty Contention No. 8: ATWS

81 Mr. Doherty contends that ACNGS is not adequately protected against an anticipated transient without scram ("ATWS") event because ACNGS has only a manually operated SCRAM system as a redundant system.

82 The possibility of an ATWS event is currently an unresolved safety issue (A-9). This item has been the subject of extensive study by the NRC Staff. The chronology of such studies are described in Supplement No. 2 to the SER (pp. C-17 to C-19), and Supplement No. 4 to the SER states (at p. C-2) that the final Staff resolution of ATWS is contained in Volume 4 of NUREG-0460 "Anticipated Transients Without Scram for Light Water Reactors" (March, 1980). The Staff has determined that the ATWS generic issue has been technically resolved. (Staff witness Moon, p. 4, following Tr. 21233). The Commission is now considering adoption of a rule that would establish specific design criteria to mitigate against the consequences of an ATWS event. (See, 46 Fed. Reg. 57521, Nov. 24, 1981).

83 Given the foregoing facts it is no longer necessary for the Board to consider this contention since it deals with a generic issue that has become the subject of a general rulemaking. Potomac Electric Power Co. (Douglas Point Nuclear Generating Station, Units 1 & 2), ALAB-218,8 AEC 79 (1974) Nonetheless, the Board does note that there is no basis to the contention considering the presently contemplated design. In Appendix K of the PSAR, Applicant committed to incorporate any design

modifications that may be required by the Staff to resolve the ATWS issue. At the construction permit stage, HL&P is designing ACNGS to accomodate the design alternatives in NUREG-0460, Vol. 4. (Applicant's witness Robertson, p. 3; following Tr. 15526).

84 Applicant's witness, Mr. Robertson, testified that ACNGS is being designed on the assumption that it will have to incorporate, at a minimum: (1) an alternative rod injection system (ARI); (2) recirculation pump trip (RPT); (3) an automated standby liquid control system (SLCS); (4) logic changes to reduce vessel isolation events and permit feedwater runback; (5) modification of scram discharge volume to reduce common mode failure potential of the single scram discharge volume; and (6) provisions for the rapid closure of containment isolation valves in the event of fuel failure. The ARI is a redundant backup system for the prompt scram system. ARI has automatic initiation and it has completely independent circuitry. The SLCS is also a redundant system and it functions automatically. (Applicant's witness Robertson, pp. 2-3, following Tr. 15526).

85 The Staff has determined that construction of the proposed ACNGS facility can proceed without undue risk to the health and safety of the public by subsequent

operation of the facility because (1) the facility will be designed in accordance with an ATWS regulation or (2) the Staff's proposed position could be implemented before operation (Staff Exh. 21, p. C-3). The Board concurs in this conclusion and notes that it is consistent with the Commission's determination that there is reasonable assurance of safety for continued operation of existing plants pending implementation of the proposed rule. See, 46 Fed. Reg. 57521, 57522 (Nov. 24, 1981).

Doherty Contention 9: Containment Buckling

86 Mr. Doherty has alleged in Contention 9 that the ACNGS containment is not strong enough to resist dynamic and static loads that may be placed on the containment. His sole basis for this contention is a preliminary report published in January 1978, by International Structural Engineers, Inc. (ISE), one of the NRC's consultants. This report included four preliminary observations which were cited by Mr. Doherty as criticisms of the present predictive methods used for the buckling evaluation of containment vessels.

87 The four observations contained in the preliminary report cited by Mr. Doherty have been superceded in whole or part by the final report prepared by the consultant or are well accounted for in the ACNGS containment design. (Applicant's witness Mokhtarian, pp. 5-8, following Tr. 11044).

88 The first allegation raised by Mr. Doherty is that there is inadequate experimental data to determine design criteria. However, the final consultant's report shows that there is adequate data to establish criteria on axisymmetric static loads. There remains some uncertainty as to dynamic asymmetric loads, but this can be accounted for through the use of conservative techniques in the design. (Applicant's witness Mokhtarian, p. 6, following Tr. 11044; Staff's witness Chan, p. 5, following Tr. 11194; Tr. 11064-74). Second, Mr. Doherty alleged that the use of computer programs for determining buckling loads could result in inaccuracies. However, the inaccuracies are known and are accounted for through knockdown and plasticity reduction factors. The final report by ISE recommends use of this technique. (Applicant's witness Mokhtarian, p. 7, following Tr. 11044; Tr. 11079-81; 11124-26). Third, Mr. Doherty alleged that the ASME Code allows designers

to choose the least conservative method of calculating buckling stress. However, Applicant will use the approach recommended by ISE. This approach is widely used and is conservative. (Applicant's witness Mokhtarian, p. 7, following Tr. 11044; Tr. 11126-27). Finally, Mr. Doherty alleges that the NRC's consultant has recommended that a safety factor of 3 should be used because of the alleged inadequacies described above. However, the final consultant's report concludes that a safety factor of 2, as used for the ACNGS design, is adequate. This is consistent with the recommendations of the ASME Code. (Applicant's witness Mokhtarian, pp. 7-8, following Tr. 11044; Tr. 11081-84).

89 The ACNGS steel containment will be constructed in accordance with the ASME Code and the design meets the NRC's present design requirements (Staff's witness Chan, p. 4, following Tr. 11194). Chicago Bridge & Iron Company (CBI) will manufacture the steel containment. CBI is also responsible for performing the required analyses and design activities. To do this task, the containment vessel is mathematically modeled using Kalnins' Shells of Revolution Program. The program accounts for axisymmetric and nonaxisymmetric loads on axisymmetric shell structures. Conservative

assumptions are made to account for dynamic loads and reduction factors are employed in the assessment of structural stability. (Applicant's witness Mokhtarian, pp. 2-5, following Tr. 11044). The method of analysis employed for the design of the ACNGS containment vessel will result in a conservative prediction of stresses and the buckling evaluation method employed will produce a safe and conservative design. (Applicant's witness Mokhtarian, p. 8, following Tr. 11044; Tr. 11224-27). Indeed, the Applicant has chosen to use the most conservative of the three alternative methodologies mentioned in the relevant portion of the ASME Code. (Staff's witness Chan, Tr. 11235).

90 Containment buckling is on the Staff's list of generic technical activities. It is not anticipated that the Staff's work in this area will result in significant design changes, but rather will produce more definitive requirements for future licensing activities (Staff's witness Chan, pp. 2-5, following Tr. 11194; Tr. 11197-8, 11213-22). The Staff has determined that ACNGS complies with Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components." (Staff's witness Chan, p. 4, following Tr. 11194).

91 Despite the fact that containment buckling is a generic technical activity, the Staff has determined licensing of ACNGS does not endanger the health and safety of the public. This conclusion is based on the Staff's determination that the steel containment for ACNGS is designed for the loads which may give rise to buckling. The Staff has determined that the conservatism associated with the loads on the containment compensate for uncertainties related to buckling calculations. Also, the Staff believes that compliance with Reg. Guide 1.57 provides an ample margin of safety. (Staff's witness Chan, pp. 5-6, following Tr. 11194). Moreover, the stiffeners used in the ACNGS steel containment will reduce the potential for buckling due to geometrical imperfections in the shell. If the Staff's research program on this subject shows that additional strengthening of the containments is required, this can be done by the addition of more stiffeners. (Id., Tr. 11099-101).

92 Based on the foregoing findings, the Board finds that there is no merit to Mr. Doherty's contention that the ACNGS containment is not strong enough to resist dynamic and static loads.

Doherty Contention No. 10: Diesel Generator Reliability

93 In Contention 10, Intervenor Doherty alleges that the diesel generator for the ECCS high pressure core spray system (HPCS) and the other plant diesel generators are not reliable in start-up and operation. Mr. Doherty cites instances of past failures of diesel generators, and refers to an NRC contractor report "Enhancement of On-Site Emergency Diesel Reliability" (NUREG/CR-0660) in support of his contention.

94 NUREG/CR-0660 is a document which offers recommendations by an NRC contractor to improve diesel generator reliability. The Applicant provided detailed testimony with respect to each of the recommendations in NUREG/CR-0660 and the manner in which Applicant intended to meet the recommendations. (Applicant's witnesses Clarke and Montalbano, pp. 3-10, following Tr. 14716). This testimony covered the HPCS diesel generator which is within General Electric's scope of supply, and the Division 1 and 2 safety diesel generators which are being provided by Ebasco. The NRC Staff stated that prior to the issuance of the operating license for ACNGS, it will independently determine which of the recommendations made in NUREG/CR-0660 should be applied

to ACNGS, and will assure that ACNGS meets those requirements. (Staff's witnesses Chopra and Giardina, p. 3, following Tr. 14646).

95 The Applicant's testimony reflects the commitments which Applicant has made to meet the recommendations of NUREG/CR-0660:

1. Provide an air dryer, either dessicant or refrigerant type, which will reduce moisture in the air and thereby prevent the condensation of water vapor in the air starting system and insure that the diesel generator will start when called upon. (Applicant's witnesses Clarke and Montalbano, pp. 3-4 following Tr. 14716).

2. Provide dust tight enclosures surrounding instruments, relays, control switches, etc., and provide washable filters around louvered cabinets where required. (Id., p. 4).

3. Provide a turbocharger heavy duty drive assembly which can withstand the required no load and light load operations. (This applies only to the HPCS diesel and not to the Division 1 and 2 safety diesels). (Id., pp. 4-5).

4. Provide training to insure that maintenance personnel are aware of maintenance procedures. (Id., p. 6).

5. Provide a continuous lube oil circulating system for both the HPCS and Division 1 and 2 safety diesels. (*Id.*, p. 6).

6. Perform testing procedures and preventive maintenance procedures in accordance with NRC requirements and recommendations of the manufacturer. (*Id.*, p. 7).

7. Train HL&P operating personnel to determine and correct the "root cause" of diesel generator problems. (*Id.*, p. 7).

8. Reduce the amount of dust and contaminants in and around the diesel generators by implementing certain recommendations relating to the intake and exhaustion of air for the diesels. (*Id.*, pp. 7-8).

9. Provide a gravity drain for the ACNGS bulk fuel tanks. (*Id.*, p. 8).

10. Provide generator protection during overload by specifying an electrical insulation which is able to withstand the temperature rise associated with an extended overload event. (*Id.*, pp. 8-9).

11. Maintain cooling water temperature control for the diesels. (*Id.*, p. 9).

12. Paint the diesel generator floors. (*Id.*, p. 9).

13. Locate engine/generator control panels as specified by the manufacturer. (Id., pp. 9-10).

In addition to the above commitments, Applicant has taken into account, in the ACNGS diesel generator design, certain NRC inspection and enforcement circulars (Nos. 79-12 and 79-23) which relate to the operation of the diesel generators.

96 The Board finds that, contrary to Mr. Doherty's allegations, implementation of the recommendations of NUREG/CR-0660 will assure the reliability of the diesel generators. Further, the Board is satisfied that implementation of both the Applicant's commitments and the Staff's qualification and preoperational testing program will provide sufficient interim assurance. Staff will review Applicant's implementation of the NUREG/CR-0660 recommendations at the OL stage of review and will require any necessary changes at that time. The Staff's qualification and preoperational testing program will require 300 tests with no more than 3 failures if the diesels are prototypes and 69 tests before operation when the diesels are installed in the plant. (Staff's witnesses Chopra and Giardina, pp. 3-4, following Tr. 14646; Tr. 14847). In addition, there

will be periodic in-service testing required of the diesels. (Staff's witnesses Chopra and Giardina, p. 4, following Tr. 14646). Accordingly, the Board finds that there is no merit to Mr. Doherty's Contention 10.

Doherty Contention No. 11: Spent Fuel Pool Accident

97 In contention 11, Intervenor Doherty claims that a spent fuel pool handling accident has not been adequately considered with respect to a spent fuel assembly being dropped onto the spent fuel pool floor.*/

98 The floor of the spent fuel pool consists of a 1/4 inch thick stainless steel liner, with a 6 ft. thick reinforced concrete mat beneath the liner. Both Applicant and Staff performed analyses which assumed that a spent fuel assembly was dropped onto the floor of the spent fuel pool. These analyses, which did not take credit for the concrete floor beneath the pool, indicated that the spent fuel pool liner will not be penetrated from the dropped fuel assembly. (Applicant's witness Chiou, p. 2, following Tr. 19509; 19521-22; Staff's witness Leung, pp. 2-3, following Tr. 20736; 20741-42).

---

*/ As stated by the Licensing Board in its Second Order Ruling Upon Motions for Summary Disposition, September 1, 1981, pp. 33-34.

99 Despite the fact that a dropped assembly will not penetrate the pool liner, Applicant and Staff further analyzed the loss of water inventory in the spent fuel pool due to a postulated breach at a weld in the pool liner. The Applicant's calculations indicated that loss of water inventory through the normally closed leakage detection system valve as a result of the breach at a weld in the pool liner would be limited to less than 39 gpm. (Applicant's witness Malec, pp. 4-5, following Tr. 19509). The Staff calculated a leakage of less than 75 gpm. (Staff's witness Leung, p. 4, following Tr. 20736). Neither calculation presents a potential safety problem since the Essential Service Cooling Water System can provide approximately 100 gpm of makeup water from either of its trains to the spent fuel pool. The non-safety related demineralized water system can provide an additional 50 gpm of makeup water. (Applicant's witness Malec, p. 5 following Tr. 19509; Staff's witness Leung, p. 4, following Tr. 20736).

100 Finally, assuming a hypothetical penetration of the spent fuel pool liner by a dropped fuel assembly, no unacceptable radiological consequences will occur. Assuming that all of the rods in the dropped fuel

assembly would be damaged, radioactivity releases could escape either (1) through the punctured liner or (2) through the pool surface. The first pathway, e.g., the punctured liner, is not considered reasonable because water leakage through the damaged liner would be collected by the pool liner leak detection system, and then routed to the radioactive waste treatment system. If the liner were damaged in an area not serviced by this system, the 6 ft. thick concrete beneath the pool liner would limit any potential leakage of radioactivity from the water in the pool. (Applicant's witness Martin, pp. 2-4, following Tr. 19509). The most likely pathway for release of radioactivity would be through the fuel handling building. However, the consequences from such releases are bounded by the analysis for the design basis spent fuel pool accident in Chapter 15 of the PSAR which assumes that 98 fuel rods fail, whereas a maximum of 62 fuel rods could be damaged as a result of a drop of one fule assembly onto the spent fuel pool floor. The radiological doses from releases of one fuel assembly were calculated to be only a small fraction of Part 100 dose limits and well within NRC guidelines for a fuel handling accident. (Applicant's witness Martin, pp. 3-4 following 19509; Tr. 19545).

101 The Board finds that, contrary to Intervenor's claim, a dropped spent fuel assembly will not penetrate the spent fuel pool liner. In any event, even if such penetration were to occur, the consequences are not significant.

Doherty Contention 15: WIGLE Code

102 In Contention 15, Intervenor Doherty contends that the Applicant's analytical method to determine scram reactivity function for the design basis power excursion accident will underpredict the energy yield of this power excursion similar to the underprediction of the WIGLE code.*/

103 First, it is clear that the Applicant does not use the WIGLE code in any analysis for scram reactivity. (Staff's witness Brooks, p. 2 following Tr. 20948). Moreover, for the analysis of scram reactivity for the design basis power excursion accident, which for ACNGS is the rod drop accident, the Applicant used a three dimensional static code as described in NEDO-10527.

---

*/ Applicant filed a Motion for Summary Disposition on this issue which was denied by the Board in its Second Order Ruling Upon Motions For Summary Disposition dated September 1, 1981, for the reasons set forth on pages 36-39.

(Staff's witness Brooks, p. 3, following Tr. 20948; Applicant's witness Congdon, Tr. 20029-30). This code in no way is comparable to the one dimensional WIGLE code cited by Mr. Doherty in his contention. (Staff witness Brooks, p. 3, following Tr. 20948; Applicant's witness Congdon, Tr. 20034). The Applicant uses a one dimensional, time-dependent, spatial calculation (ODYN Code) to generate scram reactivity in the point-kinetics plant <u>transient</u> analyses, but it uses the three dimensional static code to generate scram reactivity in the control rod drop <u>accident</u> analysis. The static code used in this manner will underpredict the reactivity inserted by scram, and, thus, is conservative (Staff's witness Brooks, p. 3, following Tr. 20948; Tr. 20952; Applicant's witness Congdon, pp. 2-3; following Tr. 20034). */

104 In addition, the SPERT tests which Mr. Doherty cites to show that the WIGLE code underpredicted scram reactivity, bears little resemblance to a modern BWR core. For example, with the SPERT tests, no control rods were inserted, so that these tests did not measure the effects of scram reactivity. Applicant's witness testified that in his opinion the SPERT tests were so

---

*/ The adequacy of the ODYN code is discussed in the findings on Doherty Contention No. 21.

far removed from the typical BWR scram conditions that it would not be appropriate to use these tests for the assessment of conservatism of Applicant's transient analysis code (ODYN code) or even the WIGLE code for that matter. (Applicant's witness Congdon, pp. 3-4, following Tr. 20028).

105 Since the Applicant does not use a one dimensional code, similar to the WIGLE code, to calculate scram reactivity for the design basis power excursion accident, but instead uses a three dimensional static code, the Board concludes that there is no merit to Mr. Doherty's claim that the Applicant's analytical method to determine scram reactivity function for the design basis power excusion accident will underpredict the energy yield for this accident based upon the alleged inadequacies of the WIGLE code.

Doherty Contention 17: SRV Reliability

106 Mr. Doherty contends that blowdown following a Power Excursion Accident (PEA), Loss-of-Coolant Accident (LOCA) or Power Coolant Mismatch Accident (PCMA) combined with a single or several safety relief valves stuck in either the fully open or fully closed position may cause loads which would crack the containment wall. Mr. Doherty also questions the reliability of the

valves and requests that Applicant be required to use the most reliable valve available or a variety of valves from different suppliers.

107 Safety/relief valves (SRVs) protect against overpressurization of the Reactor Coolant Pressure Boundary (RCPB) by opening automatically in either the relief or safety modes of operation when the pressure setpoints are exceeded. ACNGS has 19 safety/relief valves. The present design of the system is such that the 19 SRV's open at different pressure levels via the relief function. At 1103 psig, 1 valve opens; at 1113 psig, 9 more valves open; at 1123 psig, the remaining 9 valves open. In the relief mode the valves are opened automatically and can also be operated by operator action. If the reactor pressure exceeds 1123 psig and one or more of the safety/relief valves are not open, the valves will open automatically in the safety mode of operation. (Applicant's witness Boseman, pp. 4-5, following Tr. 16146).

108 The design of ACNGS SRV's, manufactured by Crosby Valve and Gauge Company, has eliminated the causes of previously experienced undesirable performances associated principally with reverse-seated type multiple stage, pilot-operated safety relief valves. The ACNGS SRV's do not use pilot or air operator diaphragms. Instead,

the ACNGS SRV's consist of a direct-acting safety valve with an electropneumatic actuator assembly to provide for two separate and independent modes of operation (safety and relief). Crosby safety and relief valves have been used in both nuclear and non-nuclear applications for many years with an excellent service and performance record. The design of the SRV to be used at ACNGS is an improved design based upon the experience gained in over 100 reactor years of BWR operations. The improved SRV design has been used with success at the Browns Ferry III and Chinshan nuclear plants.*/ Moreover, the present SRV design has undergone extensive qualification testing. The design improvement combined with existing manufacturing control of critical dimensions and clearances between all moving parts, stringent production testing and in-service maintenance and inspection will substantially improve the reliability of the SRV's. (Applicant's witness Boseman, pp. 2-4, following Tr. 16146; Tr. 16147-66; Tr. 16211-12). In fact, the failure rate is expected to be reduced by a factor of 10 over the pilot-operated valves. (Staff's witness Hodges, p. 7, following, Tr. 15128).

---

*/ The valves have also recently undergone successful preoperational testing at the Kuosheng Nuclear Plant in Taiwan. (Applicant's witness Hucik, Tr. 20381-83).

109 Mr. Doherty alleged that the number of SRV's had been reduced from 22 to 19. Such a reduction is of no consequence. Up to one-half of the 19 valves could fail to open in the relief mode and still maintain the vessel pressure at less than the code limit of 110% of design pressure for the most severe overpressure transient. Spurious opening of an SRV is an operational nuisance, not a safety problem. Reducing the number of valves just reduces the probability of a spurious opening. Likewise, failure of a relief valve to close after opening on demand presents no safety problem. If the relief valve sticks open it will continue to heat the suppression pool but will not impart any significant dynamic loadings on the containment. For the case of a relief valve stuck open, the worst case for containment design is a combination of LOCA blowdown loads with the loads due to a single SRV actuation. The dynamic loadings due to this load case would be the same as for a LOCA plus a stuck-open relief valve. For the case of relief valves failing in the closed position, the load due to the Automatic Depressurization System (ADS) demonstrates that only 8 of the 19 relief valves are necessary to rapidly depressurize the reactor.

Thus, the failure of up to eleven relief valves in the closed position will not cause the violation of a safety limit. The pressure loadings on containment due to a LOCA plus single SRV actuation and ADS are part of the design basis for the containment as described in Chapters 3 and 6 of the ACNGS PSAR. (Applicant's witness Hucik, pp. 5-7, following Tr. 16146; Staff's witness Hodges, pp. 8-9, following Tr. 15128; Tr. 16214-15).

110 Given the foregoing facts, the Board finds that the proposed SRVs for ACNGS will be reliable, and there is no reason to require use of a variety of SRV's manufactured by different suppliers. Indeed, such a variety could lead to a reduction of reliability by adding needless complications. Moreover, the Board finds that the design basis loading combinations on the ACNGS containment are conservative and account for the impact of stuck open and failed SRV's. (Staff's witness Hodges, p. 9, following Tr. 15128; Boseman, Tr. 16215-16).

Doherty Contention 20(a): Gap Conductance

111 Doherty Contention 20(a) alleges that the Applicant has underestimated the amount of fission gas that is released from the fuel to the gap during a loss of

coolant accident (LOCA), and further alleges that this underestimation will result in higher peak clad temperatures. Mr. Doherty claims that the underprediction is caused by the fact that fission gas releases will be higher in fuel rods with a burn-up of greater than 20,000 megawatt-days per ton (MWd/t). Accordingly, Mr. Doherty asserts that Applicant should not be permitted to use fuel with a burn-up in excess of 24,000 MWd/t.

112 As fissioning occurs in the fuel pellets, fission gases are produced and most of the gases are trapped in the fuel pellets. A small fraction of the gas is released to the gap between the fuel pellets and the cladding. The amount of fission gas released is primarily a function of temperature. The release of fission gas, in turn, affects the surface temperature of the cladding through gap conductance. Gap conductance is affected by the gap size, fuel swelling, fuel relocation, fuel densification and by the thermal conductivity of the gas mixture. Fission gas release directly affects thermal conductivity, i.e., as fission gas releases increase, clad temperature increases. (Applicant's witness Holtzclaw, pp. 15-16, following Tr. 11750).

113 Applicart's PSAR analysis is based on General Electric's gap conductance model known as GEGAP III. (Applicant's witness Holtzclaw, p. 16, following Tr. 11750). There is no burn-up dependence in the GE model and for this reason GEGAP may underpredict the release of fission gas from the fuel at burn-ups in excess of 20,000 MWd/t. (Staff's witness R. Meyer, p. 4, following Tr. 14019; Tr. 11975; 14188-90). However, the Dutt-Baker correction factor, which is used to predict fission gas release for fuel with burn-up greater than 20,000 MWd/t, has been applied to the GEGAP model at the NRC's request. (Applicant's witness Holtzclaw, Tr. 11984). In so doing, GE has determined that even with this enhancement factor the calculated peak clad temperature is still below the 2200°F limitation imposed by 10 CFR § 50.46. (Applicant's witness Holtzclaw, pp. 17-18, following Tr. 11750; Staff's witness R. Meyer, pp. 4-5, following Tr. 14019; Tr. 11986). In fact, this correction is overly conservative (Staff's witness Meyer, Tr. 14193-96, 14200); and there is no reason to believe that use of rods with burn-up in excess of 24,000 MWd/t presents any safety concern.

114 GE has submitted a new fuel performance model, GESTR, to the NRC for review. This model has an explicit exposure enhancement factor. Once approved by the NRC,

the new model will be used by Applicant in its Final Safety Analysis Report. (Applicant's witness Holtzclaw, p. 18, following Tr. 11750). The Staff will make a final review of the Applicant's analysis at the operating license stage. However, it can now be concluded that no limit on fuel rod burn-up for ACNGS is necessary because operating limits can be adjusted, if needed, to accommodate the effects of fission gas release on fuel performance. (Staff's witness R. Meyer, pp. 5-6, following Tr. 14019; Tr. 14201-03; 14233-36). Accordingly, the Board finds that there is no basis in the record to grant Mr. Doherty's request that the Applicant be prevented from using fuel with a burn-up in excess of 24,000 MWd/t.

Doherty Contention 21: ODYN Code

<u>115</u> In Contention 21, as reworded by the Board on August 12, 1981, Mr. Doherty challenges the adequacy of General Electric's ODYN Code as (1) it applies to a reactor with a high power density like ACNGS and (2) it calculates the scram reactivity curve.

<u>116</u> The ODYN Code is a computer code developed by General Electric and is used to model the response of the

reactor to various pressure transients. The Code can be used for all types of BWR's regardless of core power density since the equations and calculations used in the Code relating to neutronic behavior are not dependent on core power density, and the transients which are sensitive to variations in core power density and other parameters depending upon the particular facility, are well accounted for in the model (Applicant's witness Huang, p. 2, following Tr. 20756, Staff's witness Brooks, p. 4, following Tr. 21084).

117 The Code has been qualified by pressure transient tests (turbine trips) at Peach Bottom 2, and the KKM reactor in Switzerland. These test results are in agreement with the Code calculations and demonstrate that the Code accurately predicts transient responses at various power densities. Moreover, preliminary indications from start-up tests at the Kuosheng 1 reactor in Taiwan indicate that ODYN Code predicted responses are conservative compared to test results (Applicant's witness Huang, pp. 2-3, following Tr. 20756).

118 The Staff has approved use of the ODYN Code and no further verification of the Code is required. However, the Staff has imposed an adder to all calculated results because of uncertainties in input parameters (Staff's

witness Brooks, p. 2, following Tr. 21084; Tr. 21096-97). This adder results in a large amount of conservatism in the Code results. (Applicant's witness Huang, p. 5, following Tr. 20756; Tr. 20762).

119 The ODYN Code has been verified as to its adequacy to generate a scram curve by comparison with solutions obtained from a three-dimensional code called PANACEA which is the code used for nuclear design. This latter code has been reviewed and approved by the Staff. The comparison, which is documented in NEDO-24154, showed that the ODYN Code calculated smaller scram reactivity and therefore was more conservative than the three-dimensional model (Staff's witness Brooks, pp. 3-4, following Tr. 21084; Applicant's witness Huang, pp. 3-4, following Tr. 20756; Tr. 20760-61).

120 The transient analyses for ACNGS using the ODYN Code will be reviewed by the Staff at the operating license stage. However, there is no indication that the results for ACNGS will lead to a derating of ACNGS. The ODYN Code has been used in GESSAR II and in operating BWR plants and in BWR/6 facilities including Kuosheng, Grand Gulf and River Bend. No derating has been necessary in these facilities and since the ACNGS design is similar to GESSAR II and these BWR/6 facilities, there

is no reason to believe that ACNGS results will prove any different. (Staff's witness Brooks, p. 3, following Tr. 21084; Applicant's witness Huang, p. 5, following Tr. 20756). Accordingly, the Board finds that while the ODYN Code analysis for ACNGS can appropriately await review at the operating license stage, there is reasonable assurance that no derating of ACNGS will be required as a result of this analysis. Likewise, the Board finds that there is no merit to Doherty Contention 21.

Doherty Contention 24: Rod Drop

121 Intervenor Doherty alleges in his Contention 24 that Applicant has not provided a basis for showing that the reactivity insertion from any dropped control rod will be sufficiently small to prevent the peak energy yield from exceeding 280 cal/gram of fuel.

122 For gross control of reactivity in the ACNGS reactor, cruciform control blades are inserted between the fuel assemblies. These control blades, or rods, enclose smaller rods filled with boron carbide, a neutron absorbing material. The reactor is controlled by driving these control blades (177 for ACNGS) into the reactor core to reduce reactivity (and thus reduce

power) and withdrawing the rods to increase reactivity (and thus increase power). There are many ways of inserting reactivity into a boiling water reactor. However, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to reactor control. It is possible, however, that a rapid removal of a high worth */ control rod could result in a potentially significant power excursion. The design basis accident dealing with rapid removal of a control rod is the rod drop accident. (Applicant's witness Stirn, Attachment RCS-1, pp. 1-2, following Tr. 20694).

123 As discussed in the findings on Doherty Contention 3, GE has determined that the peak enthalpy from a dropped rod in the worst case credible rod drop accident is a maximum of 135 cal/gm, which is well below the NRC's safety design limit of 280 cal/gm. This result was computed using an adiabatic approximation of a super-prompt, critical large core and a two-dimensional, multi-group flux representation, as discussed in "Rod Drop Accident Analysis for Large Boiling Water Reactor," NEDO-10527 (March, 1972). (Applicant's witness Stirn, Attachment RCS-1, p. 5, following Tr. 20694).

---

*/ Control rod worth is the measure of reactivity which will be added to the reactor if the rod is moved.

124 The GE methodology has been reviewed by the NRC Staff and approved for use in such analyses. This methodology has been shown to be conservative by comparisons with results obtained from a higher order calculation by Staff consultants. These results are described in BNL-NUREG-28109, "Thermal Hydraulic Effects on Center Rod Drop Accidents In A Boiling Water Reactor" (July, 1980). (Staff's witness Brooks, p. 2, following Tr. 20970).

125 Both Applicant and Staff emphasized that the rod drop accident postulated here (worst case) involves a sequence of multiple failures. (Applicant's witness Stirn, pp. 2-4, following Tr. 20694; Staff's witness Brooks, pp. 3-5, following Tr. 20970). In fact, the NRC Staff has performed a very detailed probabilistic assessment to determine the likelihood that the consequences of a rod drop event would exceed the 280 cal/gram criterion in a BWR. The Staff reached the very conservative conclusion that the probability of exceeding the 280 cal./gm. criterion was much less than $10^{-7}$ per reactor year and, hence, negligible. (Staff's witness Brooks, pp. 2-5, following Tr. 20970; Tr. 21003-04, 21069).

126 The Staff's analysis did not take credit for the sophisticated new system, known as the Rod Pattern Control

System (RPCS), to be incorporated into the ACNGS design. (Staff's witness Brooks, p. 5, following Tr. 20970). The RPCS is designed to limit the worth of any control rod so that no unacceptable effects will result from a rod drop accident. The RPCS will apply rod blocks before any rod motion can produce high worth rod patterns. The RPCS is a dual channel, safety-related system. These electronic circuits will have, in permanent storage, the identification of all rod groups and logic control information required to prevent movement of rods into unacceptable rod patterns. The RPCS, hence, limits the maximum rod worth of any rod which might uncouple and drop as discussed above. With the RPCS operational, the maximum incremental worth of any control rod is limited to approximately 0.8 percent Δ K. The Applicant testified that this very low incremental rod worth will produce a specific enthalpy well below the NRC design limit of 280 cal/gm. (Applicant's witness Stirn, pp. 3-4, and Attachment RCS-1, pp. 4-5, following Tr. 20694). The Staff concluded that this detailed probablistic assessment, coupled with the RPCS, make it virtually inconceivable that a rod drop event at ACNGS would result in a peak energy yield of 280 cal/gm. (Staff's witness Brooks, p. 5, following Tr. 20970).

127 Based on the foregoing findings, the Board concludes that Mr. Doherty's contention has no merit. The design basis control rod drop event is the worst case of reactivity insertion accident. To minimize its effect, a highly reliable Rod Pattern Control System has been designed for ACNGS which will maintain individual incremental control rod worth to less than 0.8 percent 1/W 2/K. As a result, the evidence demonstrates that a rod drop event cannot produce a specific fuel enthalpy in excess of 280 cal/gm.

Doherty Contention 25: Fuel Failure/Flow Blockage

Doherty Contention 14: Main Steam Line Radiation Monitor
Board Question 14

128 In Contention 25, Intervenor Doherty alleges that the design basis accident for a flow blockage incident is inadequate because it assumes blockage of but one fuel assembly. As a basis for this contention, he cites the blockage of two fuel assemblies at Fermi Unit 1 in 1966 and claims that flow blockage of more than one fuel assembly should be considered.

129 Although a flow blockage event has never occurred in a BWR and it is unlikely that one will occur (Staff's witness Meyer, p. 3, following Tr. 13625), the most likely mechanism for causing significant flow reduction in a BWR/6 is a foreign object. (Applicant's witness Williams, pp. 4-5, following Tr. 13381). Crud buildup would result in less severe blockage than that which would occur due to a foreign object. (Applicant's witness Williams, pp. 4-5, following Tr. 13381). If a foreign object were to cause flow reduction or flow blockage in a fuel assembly, it could occur in one of three possible ways:

(1) Blocking the orifice in the fuel support piece;

(2) Entering the fuel support piece and becoming lodged between the fuel support piece and the lower tie plate;

(3) Entering the lower tie plate and becoming lodged in the fuel assembly.

(Applicant's witness Williams, p. 5, following Tr. 13381).

130 It would be extremely difficult for a foreign object to block the fuel support piece orifice or the lower tie plate. In order to do so, an object would first have to get into the lower plenum of the reactor pressure vessel via the jet pumps. (Applicant's witness Williams,

p. 6, following Tr. 13381). Even in the highly improbable event that a foreign object were left in the lower plenum after initial reactor startup, it is likely that it would stay there because of the low flow velocities in that area. If it were swept up, it would have to travel a tortuous path between the control rod tube guide to reach an inlet orifice of the fuel support piece. These tube guides have a maximum gap diagonally of 6.096 inches and a minimum gap of 1.125 inches. (Applicant's witness Williams, pp. 6-7, following Tr. 13381). Moreover, if the object were to pass through the orifice, it would lodge on the lower tie plate or pass through holes of 0.410 inch diameter in the tie plate in order to enter the fuel channel. Even then, it would be stopped by the first spacer which would probably not significantly reduce the flow in one bundle or cause serious degradation of heat transfer conditions in other areas of the fuel assembly. (Applicant's witness Williams, pp. 7-8, following Tr. 13381). In fact, it would take a blockage of more than 98% of the flow area to possibly result in melting of the clad and fuel. (Applicant's witness Williams, p. 8, following Tr. 13381).

131 Contrary to Intervenor's allegations, there is no need to consider flow blockage of more than one fuel assembly. Unlike the design of the Fermi Unit 1 relied upon by Intervenor, ACNGS is designed with each support piece supporting four fuel assemblies; there are four separate vertical (rather than horizontal) orifices spaced at 90° intervals to allow coolant to enter each bundle. Thus, unless more than one object were to simultaneously block each of these separate flow paths, it is virtually impossible for a single object to block multiple fuel assemblies. (Applicant's witness Williams, pp. 9-10, following Tr. 13381; Staff's witness Meyer, Tr. 13793-94). It is highly unlikely that a flexible piece of material could block more than one orifice because the reactor environment would tend to destroy such flexible material. (Applicant's witness Williams, Tr. 13605-06; Staff's witness Meyer, Tr. 13793-94).

132 In any event, the Board does not consider the flow blockage of one or even more fuel assemblies to be of critical concern. As described below, if the blockage created fuel damage it would be readily detected and the reactor brought to safe shutdown. Moreover, the testimony reflects the fact that the consequences of a flow blockage event are less than the consequences of a

rod drop accident or a LOCA, both of which have been analyzed as design basis events. (Applicant's witness Horton, Tr. 13567-68, 13571, 13585, 13601; Staff's witness Meyer, Tr. 13626-27). Therefore, the Board finds that Intervenor's allegations in Contention 25 are without merit.

133 In Contention 14, Intervenor Doherty asserts that the main steam line radiation monitors (MSRLM) cannot detect rapid fuel failure, such as could occur from a postulated flow blockage event, because the monitors cannot be set low enough due to background radiation of nitrogen-16. In support of this contention, he cites incidents at Dresden Unit 3 and Three Mile Island.

134 The MSLRMs are located on the main steam lines and are designed to detect any prompt release of fission products which would result from severe rod damage. (Applicant's witness Horton, pp. 11-12, following Tr. 13381). Even though the MSLRMs operate in the presence of a normally high level of nitrogen-16, nevertheless, they would detect the failure of 1 to 3 rods which had released 100% of their fission gases, or about 150 rods which had released about 2% of their fission products. (Staff's witness Meyer, p. 2, following Tr. 13625; Tr. 13629, 13778). Once an MSLRM detects fuel failure,

isolation of the main steam line and a reactor scram is automatically initiated. This would occur approximately 15-20 seconds after fuel melting had begun. (Applicant's witness Horton, p. 12, following Tr. 13381; Staff's witness Meyer, p. 4, following Tr. 13625). While the MSLRMs are designed to detect rapid fuel failure, and to initiate a reactor scram, the offgas radiation monitoring system is designed to detect low level emissions of noble gases. The offgas system radiation monitor can detect any change in the increase of noble gases on the order of 2 to 10 curies per second; with a 2% release of fission products, it would detect the failure of a single rod in 2 to 3 minutes. Upon detection, the offgas system radiation monitor would initiate an alarm to indicate that there had been an increase in these noble gases. (Applicant's witness Horton, p. 12, following Tr. 13381; Staff's witness Meyer, p. 3, following Tr. 13625). Upon the occurrence of an offgas system radiation alarm, the operator could take necessary action including power reduction or shutting down the reactor. (Applicant's witness Horton, p. 13, following Tr. 13381).

135 The incidents at Dresden 3 and TMI 2 cited by Mr. Doherty are not relevant to this contention. First of

all, the incident at Dresden 3 was not a rapid fuel failure but resulted from a failure mechanism known as pellet-clad interaction (PCI). With PCI, fission gas releases were released over an extended period of time and were detected by the offgas system radiation monitor, not the MSLRMs. (Applicant's witness Horton, p. 14, following Tr. 13381; Staff's witness Meyer, Tr. 13627-29). Secondly, TMI 2 is a PWR, and since the coolant from the reactor of a PWR is not mixed with the steam used to drive the turbine, radiation monitors on the steam lines or in the condenser air evacuation system would not detect fuel failures as it would in a BWR. (Applicant's witness Horton, p. 14, following Tr. 13381).

136 While NUREG-0401 states that the MSLRMs cannot detect the early stages of a severe flow blockage event, the testimony reflects that there is no reason for the MSLRMs to be set lower or to act faster. (Staff's witness Meyer, Tr. 13644, 13655-56). The purpose of these monitors is to respond to large releases of fission products which is an indication of gross failure of the fuel rods. (Applicant's witness Horton, Tr. 13400). The offgas radiation monitors will detect and alarm when lower amounts of fission products are released. (Staff's witness Meyer, p. 3, following

Tr. 13625). The Board finds that the MSLRMs and the offgas radiation monitors will adequately detect both rapid and slowly progressing fuel failures, and accordingly, will provide protection for the public health and safety.

137 In response to Board Question 14, the Staff and Applicant both testified that the PCI failures at Dresden 3 were detected by the offgas radiation monitors, and that if similar failures occurred at ACNGS, they too would be detected by the offgas system radiation monitors and appropriate operator action could be taken. (Applicant's witness Horton, Tr. 13391-93; Staff's witness Meyer, Tr. 13627-31). Accordingly, the Board finds that no safety hazard would be presented even if such an event occurred at ACNGS. Moreover, there are now administrative controls that limit the rapid movement of control rods, so it is unlikely that a similar event would be repeated at ACNGS. (Applicant's witness Williams, Tr. 13420-21, 13424, 13426-27).

Doherty Contention 26: Stud Bolts

138 In contention 26, Intervenor Doherty alleges that the stud bolts for the ACNGS reactor vessel will not meet the minimum standards for yield strength during an

anticipated transient without scram (ATWS).

139 The applicable criteria for the design of the ACNGS stud bolts is set forth in 10 CFR §50.55a and 10 CFR Part 50, Appendix A, Criterion 30. These sections incorporate the requirements of Division 1, Sections III and XI of the ASME Boiler and Pressure Code. In addition, Regulatory Guide 1.65 supplements these requirements. (Applicant's witness Call, pp. 2-3, following Tr. 15655). A commitment by the Applicant in the PSAR to comply with this stated criteria is sufficient for purposes of Staff review at the construction permit stage. (Staff witness Moon, p. 2, following Tr. 20910). In fact, Applicant testified that the stud bolts for ACNGS, which have already been fabricated, meet all of the requirements of the ASME Code and Regulatory Guide 1.65. (Applicant's witness Call, p. 7, following Tr. 15655).

140 The stud bolts are made of Chromium-Nickel-Molybdenum alloy steel as specified by the ASME Code. (Applicant's witness Call, p. 3, following Tr. 15655). There are certain specifications, such as mechanical properties, chemical content, and heat treatment, which are imposed upon the manufacturer of this material. (Applicant's witness Call, p. 3, following Tr. 15655). In

addition, tests are performed on each "lot" of the stud bolt material to determine whether the material properties are adequate. (Applicant's witness Call, pp. 3-4, following Tr. 15655). In addition, the stud bolts are coated with a manganese phosphate coating to prevent corrosion, as provided for in Regulatory Guide 1.65. (Applicant's witness Call, p. 4, following Tr. 15655). Finally, the ACNGS pressure vessel has been tested to 1.25 times design pressure (or 1600 psig) and no defects were found. This demonstrates that the stud bolts can withstand severe static overload conditions. (Applicant's witness Call, p. 5, following Tr. 15655).

141 When the Commission adopts a final ATWS rule, Applicant will have to demonstrate that ACNGS can comply with the final provisions of the rule, including the ability of the stud bolts to withstand these ATWS stresses. (Staff witness Moon, pp. 2-3, following Tr. 20910).*/

142 The Board finds that the stud bolts will be designed in accordance with all the applicable code and NRC require-

---

*/ The Applicant has noted that GE has included an event equivalent to ATWS in the reactor design. According to the analysis, the maximum bolt stress for this event is 64 KSI whereas the ASME Code limit is 98 KSI. The average service stress for this ATWS event is 41.4 KSI, whereas the ASME Code limit is 72.6 KSI. Thus, according to GE, these ATWS stresses are within the ASME Code allowables and considerably below miminum yield strength. (Applicant's witness Call, p. 6, following Tr. 15655).

ments, including the stresses resulting from an ATWS event. A commitment to meet the applicable criteria is sufficient in accordance with Staff review procedures for acceptance at the CP stage. (Staff's witness Moon, p. 2, following Tr. 20910). Although GE has already fabricated the stud bolts and has determined that they satisfy all applicable criteria, the Staff has not reviewed GE's analyses. The Staff's review will be done at the OL stage (Ibid.) Accordingly, the Board concludes that the integrity of the stud bolts is assured because the Applicant meets the acceptance criteria of Sections III and XI of the ASME Code, Reg. Guide 1.65 and 10 CFR 50, Appendix G. The Staff review at the OL stage will assure that the stud bolts comply with all applicable requirements, including ATWS if necessary, for yield strength.

Doherty Contention 27: Reactor Pedestal

143 In Contention 27, Mr. Doherty alleges that the reactor pedestal concrete at ACNGS may be weakened by heat from a power excursion or loss of coolant accident, thereby making the restart and further operation of the reactor dangerous due to the possibility of reactor movement

attributable to thermal damage to the reactor pedestal. As support for his contention, Mr. Doherty cites incidents that have taken place at Dresden Units I and II, Three Mile Island Unit II and the SL-1 experimental reactor. Mr. Doherty contends that the reactor pedestal was damaged in each of these incidents and that the ACNGS reactor pedestal could be similarly damaged.

144 The incidents cited by Mr. Doherty are not relevent to the design of reactor pedestal proposed for ACNGS. The SL-1 reactor is a stationary, low-power test reactor with an altogether different support mechanism from that found at ACNGS (Applicant's witness Simpadyan, p. 4, following Tr. 11150). Moreover, both the Dresden and TMI facilities utilize a reinforced concrete pedestal while a steel reactor pedestal, as described below, will be used at ACNGS (Applicant's witness Simpadyan pp. 4-5, following Tr. 11150).

145 The ACNGS pedestal will consist of two concentric steel cylinders having diameters of approximately 20 and 32 feet, respectively, which are joined together by vertical and horizontal steel plates. The cylinders will be anchored to the foundation mat at the bottom. The annular space between the cylinders will be filled with concrete but such concrete will not have a load-bearing

function; rather the fill concrete is used to add mass to the reactor pedestal in order to assure dynamic responses compatible with those exhibited by the reactor vessel. (Applicant's witness Simpadyan p. 2, following Tr. 11150; Tr. 11181-82; Staff's witness Chan, pp. 6-7, following Tr. 11194; Tr. 11230). Since the fill concrete in the ACNGS reactor pedestal has no load-bearing function, any weakening of the concrete as a result of thermal stresses would have no effect on the structural integrity of the pedestal. (Applicant's witness Simpadyan pp. 3, 5, following Tr. 11150; Tr. 11185; Staff's witness Chan, pp. 6-7, following Tr. 11194; Tr. 11233).

146 Since the reactor pedestal to be used at ACNGS is steel rather than concrete, the Board finds that there is no basis for Mr. Doherty's contention.

Doherty Contention 28: Control Rod Ejection Accident

147 In Contention 28, Intervenor Doherty alleges that the Applicant has not adequately considered the consequences of a control rod ejection accident, which he claims will be more severe in terms of increased reactivity than the design basis rod drop accident. The

contention expresses concern that the control rod drive system could break loose from the reactor vessel causing the control rod to be driven out of the reactor more rapidly than that which would occur in a rod drop accident.

148 The evidence demonstrates that a postulated control rod ejection accident would not result in more rapid reactivity insertion than that caused by the rod drop accident. First of all, it is highly unlikely that such an event would occur.*/ Neither containment pressure nor pressure from the scram discharge volume (SDV) will act to drive the control rod out of the core. Pressure in the SDV is at atmospheric pressure, except after a scram, and this pressure would be nowhere near the pressure required to move any of the 177 CRD pistons. (Applicant's witness Peterson, p. 6 of Attachment DLP-1, following Tr. 18997). After a scram, the SDV is pressurized to equal reactor pressure, which means that the pressure above the CRD piston is equal to reactor pressure. Thus, there would be no additional pressure in the SDV to drive the control rods out of

---

*/ No control rod ejection event has ever occurred in a BWR. (Staff witness Hou, p. 2, following Tr. 19061).

the core. (Applicant's witness Peterson, p. 6 of Attachment DLP-1, following Tr. 18997). In addition, there are check valves in the 177 lines leading to the SDV, thereby precluding any transmission of excess pressure back to the CRD unit. (Applicant's witness Peterson, p. 6 of Attachment DLP-1, following Tr. 18997).

149 Secondly, containment pressure will not produce a control rod ejection event, as alleged by Mr. Doherty. The CRD units are sealed, which means that unless the system were ruptured, the drive water would not be affected by containment pressure. Only if there were a system rupture would containment pressure have any effect; in this case, the effect would be to relieve the pressure above and below the drive piston. The result of the release of this pressure would drive the control rods into, not out of, the core. (Applicant's witness Peterson, p. 7 of Attachment DLP-1, following Tr. 18997). Even if pressure below the drive piston were relieved by some incredible failure, no control rod withdrawal would occur because the CRD collet latch mechanism would hold the control rod in place. This latch mechanism requires at least 100 psi above reactor pressure in order to unlatch. Even in the unlikely event that the collet mechanism were assumed to be

jammed, the control rod withdrawal velocity would only be 2 ft./sec compared with the 5 ft./sec for a rod drop event. Thus, this event would not produce an event comparable to the rod drop event. (Applicant's witness Peterson, pp. 7-8 of Attachment DLP-1, following Tr. 18997).

150 Intervenor Doherty postulated that the entire control rod drive and housing unit might become detached from the reactor vessel, which would lead to a control rod ejection event. Such an event has never occurred. (Applicant's witness Peterson, Tr. 19010), and due to the strength and toughness of the materials, it is very unlikely that the CRD housing could become completely detached. (Staff witness Hou, p. 3, following Tr. 19061). Even if such a complete detachment were assumed to occur, the control rod cannot be ejected more than 3 inches because the support structure located underneath the CRD housing units would prevent the housing from falling more than 3 inches. A 3-inch drop of any rod in the reactor will not constitute a safety problem and is bounded by the design basis rod drop accident. (Applicant's witness Peterson, p. 2 and pp. 7-8 of Attachment DLP-1, following Tr. 18997; Staff's witness Hou at 3, 4, following Tr. 19061).

151 If a CRD housing is postulated to break-away when its rod of relative high worth is unlatched (a small fraction of total drive operation time), the rod will withdraw to the next notch position (6 inches) and, that distance added to the 3-inch drop with the housing, will result in a total withdrawal of 9 inches. Again, this withdrawal will not constitute a safety problem and is bounded by the design basis rod drop accident. (Staff's witness Hou, p. 4, following Tr. 19061).

152 The Applicant analyzed an accident whereby it was postulated that the CRD housing failed while the control rod was being withdrawn and the collet stayed unlatched. In this scenario, the control rod would continue to withdraw after the CRD housing had been stopped by the drive housing support. The analysis indicated that the equivalent steady-state rod withdrawal velocity for this occurrence would be 0.3 ft/sec. as compared with 5 ft./sec. for the rod drop event. Accordingly, this postulated event is also bounded by the design basis rod drop accident. (Applicant's witness Peterson, pp. 9-10, following Tr. 18997).

153 Finally, the Big Rock Point reactor, cited by Intervenor Doherty in his contention, is not comparable to ACNGS because the reported cracking there was in an area which has been eliminated in the ACNGS design. (Applicant's witness Peterson, p. 2, following Tr. 18997).

154 The Board finds that the above-described postulated control rod ejection events would not result in greater reactivity insertion than that which has been analyzed by the Applicant and the Staff for a rod drop accident. Accordingly, we conclude that Applicant and Staff have adequately considered the consequences of a control rod drop accident in their respective safety analyses.

Doherty Contention 29 - Blockage of Intake Canal

155 In Contention 29, Intervenor Doherty claims that postulated failure of unspecified structures could lead to unacceptable blockage of the submerged intake canal. This submerged intake canal connects the 50 acre ultimate heat sink (UHS), which is formed by excavating the floor of the cooling lake to a depth of 8 feet, to the UHS intake structure.

156 The intake canal is a safety-related reinforced concrete structure which is approximately 35 feet wide and 100 feet long. (Applicant's witness Mercurio, p. 3, following Tr. 16609). The only structure which could cause a blockage of the intake canal, if postulated to fail, is the UHS causeway which provides access to the UHS intake structure from the main plant. Even though the

UHS causeway is designed to withstand the safe shutdown earthquake, an analysis was performed in accordance with NRC Regulatory Guide 1.27 which postulated the failure of this causeway. This analysis demonstrated that soil movement resulting from a postulated failure of the causeway would be less than 4 inches. (Applicant's witness Mercurio, pp. 4-5 following Tr. 16609; Tr. 16638-39; Staff witness Pearring, p. 3, following Tr. 16692). Although movement of soil of this magnitude should not present a hazard to the intake canal, the Applicant nevertheless has committed to provide additional assurance by installing a pair of seismic Category I reinforced retaining wing walls located at the entrance to the UHS intake canal. The purpose of these wing walls, which will be angled outward at approximately 45° away from the intake canal, is to direct the postulated movement of soil material away from the intake canal, thereby restricting any soil movement from blocking the intake canal. (Applicant's witness Mercurio, p. 5, following Tr. 16609). The commitment to install the wing walls has satisfied the Staff's concerns stated in Section 2.5.4 of Supplement 2 to the SER. The Staff has independently determined that the intake canal will not be blocked from a postulated failure of the UHS causeway. (Staff's witness Pearring, p. 3, following Tr. 16692).

157 Finally, to alleviate a Staff concern (See Staff Exh. 19, SER Supp. 2, p. 2-18), the slope of the UHS submerged intake canal has been flattened to allow the sediment to assume its natural angle of repose so that it would not flow into the UHS intake structure. (Applicant's witness Mercurio, p. 6, following Tr. 16609; Tr. 16620). As a final measure, the Applicant will install a 1 foot high sill which will be placed at the front of the UHS intake structure. This sill will assure that any postulated sediment in the UHS will not flow into the intake structure. (Applicant's witness Mercurio, p. 6, following Tr. 16609; Tr. 16620).

158 Based on the foregoing findings the Board concludes that a postulated failure of the UHS causeway will not lead to blockage of the submerged UHS intake canal.

Doherty Contention 30: Interconnection/Grid Stability

159 Mr. Doherty alleges in Contention 30 that Applicant's lack of interconnections with electric utilities outside the State of Texas leaves ACNGS vulnerable to loss of off-site power during severe climatic conditions. Mr. Doherty claims that this increased potential for

loss of off-site power would, in turn, increase dependence on the use of diesel generators, which he also claims are unreliable.*/

160 Applicant's witness, Mr. F. J. Meyer, Jr., explained that HL&P already has five interconnections with other electric utilites in the electric grid that covers the State of Texas, and it is planned that the number will grow to eleven by the time ACNGS goes into operation. (Applicant's witness F. J. Meyer, Jr., pp. 3-6, following Tr. 14592). The grid of which HL&P is a member is known as the Texas Interconnected Systems (TIS), which in conjunction with many smaller utility and municipal systems form the Electric Reliability Council of Texas (ERCOT). Mr. Meyer explained that the entire TIS grid is tested in joint studies with other members of TIS in order to evaluate the response of the TIS system (both existing and planned) to simulated disturbances. These simulated disturbances include: (a) the loss of any two transmission lines; (b) the loss of any one transmission line and the loss of any one generator; and (c) the loss of any two generators. These studies clearly demonstrate that the TIS grid is stable because

*/ Diesel generator reliability is the subject of Doherty Contention 10, _supra_.

they show there will be no system separation and subsequent power loss anywhere in TIS for the contingencies considered. (Applicant's witness F. J. Meyer, pp. 8-12, following Tr. 14592). The contingencies considered actually exceed the requirements of 10 CFR Part 50, General Design Criterion 17. (Applicant's witness F. J. Meyer, Jr., pp. 7-13, following Tr. 14592; Staff's witness Chopra, pp. 6-7, following Tr. 14646).

161 Mr. Meyer testified that this type of joint testing was unique to the electric utilities in TIS (Tr. 14621). He further testified that the operating philosophy of TIS is also unique in that they do not separate from a member that is in trouble. Rather the members of TIS maintain their interconnections and shed load throughout the entire State of Texas until the grid is stabilized (Applicant's witness F. J. Meyer, Jr., p. 11, following Tr. 14592; Tr. 14621).

162 The Staff has examined the history of outages in ERCOT and found nothing to indicate any problem with grid stability. (Staff's witness Chopra, Tr. 14690-92; Tr. 14696-97). Unless there is a history of outages, the Staff does not undertake a detailed review of electric grid stability. (Staff's witness Chopra, Tr. 14806). Indeed, the loss of off-site power is not

critical to the Staff's ultimate conclusion about plant safety because they assume that all off-site power is lost for purposes of safe shutdown of the plant. Thus, in Staff's view the reliability of on-site power is more critical than the reliability of off-site power. (Staff's witness Chopra; Tr. 14806-12). The reliability of on-site power is the subject of Doherty Contention No. 10. As described in the findings on that contention the existing diesel generators provided adequate on-site power for safe shutdown.

163 In sum, the Board finds that there is no evidence of any grid stability problem on Applicant's system or the grid with which it is interconnected. Contrary to the allegations in the contention, there is no reason to require Applicant to shutdown during severe climatic conditions nor is there any reason to require changes in the number or operation of the diesel generators.

Doherty Contention 32: ECCS Vaporization Rate

164 Mr. Doherty alleges in Contention 32 that the GE Emergency Core Cooling System (ECCS) evaluation model underpredicts the generation of steam during ECCS injection flow after a loss-of-coolant accident (LOCA).

This contention cites an ACRS letter of March 13, 1979 concerning the Zimmer Nuclear Power Station, Unit 1 and refers to some tests at General Electric's Two Loop Test Apparatus (TLTA) facility which were conducted as part of GE's ECCS test program.

165 In 1978, the Staff raised a concern about the conservatism of part of the ECCS evaluation model based on a particular TLTA test where there was slower than anticipated depressurization with ECCS injection. The Staff was concerned that the slow depressurization may have been due to greater steam generation than was predicted by the GE vaporization model. GE reviewed the test data and determined that the slower depressurization was primarily due to the liquid entrained in the break flow. Subsequent tests confirmed that the slower depressurization rate for the tests with ECCS injection was due to increased liquid in the break flow and not due to core vaporization. (Applicant's witness Sozzi, pp. 4-5, following Tr. 15689; Staff witness Hodges, pp. 10-11, following Tr. 15128). Based on its review of this information, the Staff concluded that the GE model does not underpredict the vaporization rate. (Staff's witness Hodges, p. 11, following Tr. 15128).

166 The Board finds that there is no longer any basis for the concern expressed in Doherty Contention No. 32. The testimony is dispositive of the concern raised by the ACRS in connection with the Zimmer plant (Tr. 15714); there are no differences in channel boxes used at Zimmer and at ACNGS (Tr. 15710; 15378). Thus, the Board finds that there is no basis for requiring any revision to General Electric's ECCS model.

Doherty Contention 35: Welder Training

167 Intervenor Doherty contends, in contention 35, that the Applicant will be unable to provide safe welding of piping at ACNGS because of the shortage of trained welders. Mr. Doherty cites instances of welding problems at the South Texas Project (STP) and Comanche Peak in support of his contention. Intervenor contends that the Applicant should be required to present a program for training persons before they are allowed to weld at ACNGS.

168 Applicant's witness from Ebasco, who will have the primary responsibility for welding activities at ACNGS, testified that the welders to be used at ACNGS will be selected from available union personnel who have com-

pleted the union's welder training program. (Applicant's witness Gunther, p. 2, following Tr. 18526; Tr. 18536; 18597). More importantly, welders will not be hired to work at ACNGS until they have successfully completed the qualification testing administered by Ebasco. The qualification requirements are set forth in the latest NRC accepted edition and addendum to Section IX, Articles I and III of the ASME Boiler and Pressure Vessel Code. (Applicant's witness Gunther, p. 2, following Tr. 18526; Staff's witness Litton, p. 2, following Tr. 19771). Thus, no welders will be allowed to work at ACNGS until they have been fully qualified. (Staff's witness Litton, p. 5, following Tr. 19771).

169 Ebasco will employ a staff of experienced "hands on" welding specialists who will be designated as a "Special Processes Group." This Group will monitor all site welding activities, including evaluating weld defect problems and initiating action for resolution of these problems. (Applicant's witness Gunther, Tr. 18533; 18559-60; Applicant's witness Frazer, Tr. 18561-62). The Group will also monitor welding records to determine if individual welders should be removed from the job and given additional training because of an excessive number of defective welds. (Applicant's witness

Gunther, Tr. 18534; 18564-65). It should be noted that a similar welding control group used by Ebasco at the Waterford nuclear project resulted in holding the weld rejection rate in piping systems to a low level. (Applicant's witness Gunther, p. 6, following 18526).

170 HL&P, through its QA/QC personnel and organization, will oversee the implementation of the qualification testing of welders at ACNGS. (Applicant's witness Frazer, p. 8, following Tr. 18526; Tr. 18540; Staff's witness Litton, p. 4, following Tr. 19771). This oversight function will involve surveillance of welder qualification activities and will also include the monitoring and surveillance of the Ebasco QA/QC program. Lastly, HL&P will conduct comprehensive audits of the entire welding program carried out by Ebasco. (Applicant's witness Frazer, pp. 8-9, following Tr. 18526; Tr. 18540; 18547).

171 With respect to certain welding problems incurred at STP, the principal cause of these problems was identified as Brown & Root's policy of hiring welders with only rudimentary skills, putting them through a training program, qualifying them according to the requirements of the ASME Code, and then deploying them to the field. While this practice did not violate NRC regula-

tions, it did raise some practical problems as to the performance of the individual welder once he was deployed to the field. (Applicant's witness Frazer, Tr. 18528). In contrast to this practice, as discussed above, Ebasco will hire only people who hold union cards and who have had prior training in welding. (Applicant's witness Frazer, Tr. 18529-30). Moreover, Ebasco's Special Processes Group will monitor the entire welding activities, including individual performances, to insure that all welding activities are performed in accordance with project specifications. (Applicant's witness Frazer, Tr. 18531-32). Taking into account Ebasco's welder program as described above, as well as Ebasco's past record of constructing numerous nuclear plants, the Board is satisfied that the proposed welder program for ACNGS provides assurance that problems similar to those that occurred at STP will not occur at ACNGS.

172 Finally, the Applicant does not anticipate any shortages of trained welders during construction of ACNGS. There has been no shortage of welders at other HL&P power plant projects. If such a shortage did occur, qualified welders could be hired from unions outside the Houston area, or, if necessary, Ebasco could insti-

tute an appropriate training program. (Applicant's witness Frazer, Tr. 18549-50; Applicant's witness Gunther, p. 10, following Tr. 18526; Tr. 18550-51).

173 The Board finds that there is no merit to Intervenor's allegation that unsafe welding will occur at ACNGS due to a lack of qualified welders.

Doherty Contention 38(b): Cold Shutdown in 24 Hours

174 In Contention 38(b), Intervenor Doherty alleges that the ACNGS reactor cannot be brought to cold shutdown in 24 hours.*/

175 There is no NRC requirement that the reactor be brought to cold shutdown in 24 hours (Staff's witness Hodges, pp. 1-2, following Tr. 17915). Moreover, the testimony reflects that ACNGS can be brought to cold shutdown in less than 10 hours after scram. (Applicant's witness Mitchell, p. 4, following Tr. 17891; Staff's witness Hodges, p. 2, following Tr. 17915). "Cold shutdown" is reached when the reactor is subcritical and the reactor coolant temperature is less than 212°F at atmospheric

---

*/ Since the provisions of NUREG-0718 for plants in the construction permit stage plants do not require that the reactor be brought to cold shutdown within 24 hours as suggested by NUREG-0578, the Board struck that portion of the contention which referred to NUREG-0578, leaving the contention to read simply that the reactor cannot be brought to cold shutdown within 24 hours. (Tr. 17599).

pressure. Initially, after scram, the reactor is cooled down by continuing to dump steam generated in the reactor vessel to the main condenser. It will take approximately 2 hours to cool the reactor to a temperature of 344°F using the main condenser. (Applicant's witness Mitchell, p. 2, following 17891; Tr. 17892; Staff's witness Hodges, p. 2, following Tr. 17915). When the pressure decreases so that dumping steam to the condenser can no longer take place, the reactor is then cooled by placing in operation the residual heat removal (RHR) system. Prior to initiation of the RHR system, that system is flushed with reactor grade water which takes an additional 2 hours. (Applicant's witness Mitchell, p. 3, following Tr. 17891). Once the RHR system is in operation, the reactor coolant can be brought to a temperature of less than 212°F in a total of less than 10 hours (Applicant's witness Mitchell, p. 4, following Tr. 17891; Staff's witness Hodges, p. 2, following Tr. 17915). This time period could still be met even if only one loop of the RHR system were operating. (Applicant witness Mitchell, Tr. 17901).

176 As an alternate shutdown mode, suppression pool water can be injected directly into the reactor pressure vessel, and if this mode is used, the time to reach a coolant temperature of less than 212°F is significantly less than the normal mode of using the RHR system. (Applicant's witness Mitchell, pp. 4-5, following Tr. 17891).

177 The Board finds that Mr. Doherty's contention has no merit. The ACNGS reactor is capable of being brought to cold shutdown within 24 hours.

Doherty Contention 39: Fuel Swelling/Rupture

178 Doherty Contention 39, as reworded by the Board, alleges that the Applicant had not provided an adequate showing that the degree of fuel swelling and incidence of rupture are not underestimated.

179 The wording of this contention comes directly from the language of 10 CFR Part 50, Appendix K, Sec. 1.B, which describes requirements and acceptable features of evaluation methodologies. The methodology used by GE, which Applicant relies upon to demonstrate that the reactor design complies with 10 CFR §50.46, is described in "General Electric Company Model for Loss-of-Coolant

Analysis in Accordance with 10 CFR Part 50, Appendix K, NEDO 20566." (Applicant's witness Williams, p. 9, following Tr. 11750).

180 GE has determined that a Loss-of-Coolant Accident (LOCA) is the most severe accident in terms of swelling and rupture because the LOCA event causes the largest differential pressure across the cladding, and also causes the highest cladding temperature. Depressurization of the reactor pressure vessel (RPV) following a LOCA can result in the fuel rod internal pressure exceeding the RPV pressure, at which point the fuel cladding begins to deform outward. GE's assessment of the LOCA event shows that stresses due to the internal pressure of the fuel rods is below the ultimate strength of the clad and, therefore, perforations will not occur. (Applicant's witness Williams, pp. 13-14, following Tr. 11750). Furthermore, the full scale bundle tests performed by GE demonstrate that the amount of swelling which does occur during a LOCA event is not sufficient to reduce the coolability of the bundle (Applicant's witness Williams, pp. 10-14, following Tr. 11750; Tr. 12013-16).

181 The Staff has approved the GE model, subject to two revisions. Subsequent to the submission of Dr. Meyer's pre-filed testimony, General Electric did submit a revised analysis taking into account the revisions requested by the Staff. The Applicant's witness testified that the new analysis demonstrates that the GE model complies with Appendix K (Applicant's witness Williams, Tr. 11751-52). The NRC has not completed its review of the new GE analysis. Dr. Meyer testified that a final review of the ACNGS ECCS analysis will be made at the operating license stage. However, based on its prior review of the GE model and its review of other plants, Dr. Meyer anticipates that the Applicant will be able to demonstrate compliance with 10 CFR §50.46 using analytical models that meet the requirements of Appendix K. (Staff's witness R. Meyer, pp. 7-8, following Tr. 14019). This conclusion is based upon the margins to LOCA limits which are indicated in the present ECCS analysis (Staff's witness R. Meyer, p. 8 following Tr. 14019). Even if the evaluation ultimately demonstrates that these margins are not present, the LOCA criteria of §50.46 could be met by a slight reduction in peaking factor without any change

to plant design. (Staff's witness Meyer, p.8, following Tr. 14019). Therefore, the Board finds that Mr. Doherty's contention has no merit, since there is reasonable assurance the ECCS analysis for the ACNGS facility will not underestimate the degree of swelling and incidence of rupture of the cladding, and will thus comply with 10 CFR §50.46 and Appendix K.

Doherty Contention 40: Part 100 Releases.

182 In his Contention 40, Mr. Doherty alleges that "the Allens Creek site is unsuitable for the proposed nuclear plant, because the assumed fission product release from any accident considered credible will exceed the limitations of radioactivity dose to the low population zone stated in 10 C.F.R. 100.100(a)(1), (2) and (3)." Mr. Doherty alleges that the release of radioactivity at Three Mile Island was higher than would have been calculated under Regulatory Guide 1.4 and infers that since the ACNGS design is "sufficiently similar" to TMI, the releases calculated for it under accident conditions are also understated.

183 The stated basis for Mr. Doherty's contention is "Board Notification - TMI Releases (BN-79-23)" dated June 27,

1979 which contained a preliminary estimate that 13 million curies of Xenon-133 were released at TMI rather than the 600,000 curies of Xenon-133 releases that would have been calculated under the Regulatory Guide 1.4 model. Later reports estimated the releases of Xenon-133 at TMI to have been in the range of 2 to 2.5 million curies. (Staff's witness Read, pp. 2-3, following Tr. 17338).

184 Witnesses from both the NRC Staff and the Applicant testified that the higher releases of Xenon-133 at TMI were attributable to the sequence of events in that incident rather than infirmities in the Regulatory Guide model. The Regulatory Guide assumes that following a design basis accident with a high degree of core damage, fission products will be immediately released to the containment atmosphere and will then leak into the environment. This assumption is very conservative since it results in a high percentage of the shorter lived high energy emitting noble gases such as Krypton-88 being released. (Applicant's witness Martin, p. 2-4, following Tr. 16331; Staff's witness Read, pp. 3-5, following Tr. 17338).

185 At TMI, the releases to the environment did not occur instantaneously as assumed in the Regulatory Guide but

rather commenced several hours into the incident, thus allowing a significant decay time for the short-lived high energy isotopes and resulting in a higher ratio of the low energy gamma emitter, Xenon-133, being released. (Applicant's witness Martin, p. 4, following Tr. 16331). Thus, the releases from TMI were virtually all Xenon-133 which, under Reg. Guide 1.4 computations, contributes only a very small fraction of the resultant potential doses. (Staff's witness Read, p.3, following Tr. 17338).

186 Moreover, the bulk of the radioactive noble gas that was released from TMI into the environment did not leak from the containment building, but from the waste gas system in the auxiliary building. The gas in this latter system had evolved from reactor coolant water which had been transferred into the auxiliary building during the course of the accident. The Xenon in this system was radiogenic, i.e., it had been formed by the radioactive decay of iodine isotopes which were dissolved in the reactor coolant, and it escaped through the auxiliary building vent primarily during Thursday and Friday, March 29 and 30, 1979. At the time of the largest releases, most of the noble gas fission products (Xenon, Krypton) which had been in the damaged

fuel rods at 4:00 A.M. on Wednesday were in the containment building atmosphere, where they remained. (Staff's witness Read, p. 4, following Tr. 17338, Applicant's witness Martin, p. 7, following Tr. 16331).

187 Since the accident scenario at TMI was different than that assumed in Regulatory Guide 1.4, the amount of Xenon-133 actually released was different. However, these differences did not result in doses at TMI that exceeded those calculated under Regulatory Guide 1.4 or which did not meet the requirements of 10 C.F.R. Part 100. In fact, NUREG-0558 indicates that an average dose of only 1.5 millirem was received by the population surrounding TMI during the entire course of the incident. The study also indicates that the maximum estimated dose to one individual outside the exclusion area was less than 100 millirem, or 1/250th of the 10 C.F.R. Part 100 limits. (Applicant's witness Martin, p. 3, following Tr. 16331; Staff's witness Read, pp. 6-7, following Tr. 17338).

188 Mr. Doherty's inference that the releases at TMI exceeded safe levels is therefore inaccurate, as is his allegation that ACNGS is similar to TMI. First, ACNGS is a General Electric BWR with a reactor and containment design substantially different from the Babcock

and Wilcox PWR at TMI. In fact, an entirely different Regulatory Guide (Reg. Guide 1.3) is used to calculate releases for BWR'S. (Applicant's witness Martin,, p. 2, following Tr. 16331; Staff's witness Read, p. 6, following Tr. 17338). Second, ACNGS does not utilize a coolant let-down/make-up system, as provided at TMI, so a release path using this system is not possible at ACNGS. Moreover, in response to the TMI incident, design modifications were developed for the containment isolation system. These modifications are required by NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License." As detailed in Sections II.E.4.2., II.E.4.4, and III.D.1.1 of the ACNGS PSAR, Appendix O, ACNGS has incorporated all suggested modifications, such as containment isolation for non-essential systems, that were not already part of the plant design. (Applicant's witness Martin, p. 8, following Tr. 16331).

189 As a matter of interest, the Staff has estimated the effects of removing the maximum inventory of Iodine 133 at ACNGS from the containment to the auxiliary building, allowing it to decay into Xenon-133 and then immediately releasing the entire inventory to the environment. Such computation has demonstrated that it is physically

impossible to exceed the dose guidelines of 10 C.F.R. Part 100 using only radiogenic Xenon-133. (Staff's witness Read, p. 9, following Tr. 17338).

190 The Board finds that the Applicant has demonstrated that ACNGS will satisfy Part 100 dose limits. (App.Exh. 27, Table 15.1.39-3; Applicant's witness Martin, pp. 5-7, following Tr. 16331). These Part 100 dose calculations have been reviewed by the Staff in its safety evaluation. Contrary to Mr. Doherty's contention, nothing which occurred during the TMI incident reflects any doubt on that determination. In particular, it has been demonstrated by both Applicant and Staff that although the amount of Xenon-133 estimated to have been released from TMI were different that the Regulatory Guide source term, it resulted in doses that neither exceeded those calculated under Regulatory Guide 1.4 nor exceeded the requirements of 10 CFR Part 100 dose limits.

Doherty Contention 41: Reactor Water Level Indicators

191 Mr. Doherty contends that the reactor water level indicators at ACNGS are inadequate, and that Applicant should be required to install water level indicators that are redundant as to type and function. Mr. Doherty

cites as the basis for his contention events which occurred at the TMI and Oyster Creek nuclear plants.

192 After hearing the testimony on this issue, the Board concludes that there is no valid basis for this contention. First, the Oyster Creek incident involved an operator error following a false high reactor pressure scram. (Applicant's witness Lane, pp. 5-6, following Tr. 15445). Moreover, Oyster Creek is a non-jet pump BWR, and, as such, water from the reactor annulus area must pass through the recirculation piping in order to reach the reactor vessel lower plenum and the core shroud area. With the recirculation pump discharge valves closed, the only flow path back to the core region is via the 2-inch bypass lines around the discharge valves. The effect of this flow restriction was to reduce the water level in the core region and to increase the level in the reactor annulus area. This cannot happen in a BWR-6 because there is a direct pathway between the core and the annulus. In addition there were no erroneous water level indications at Oyster Creek. The instrumentation gave accurate indications of water level inside both the shroud and annulus regions. The differing water level indications that confused the operators resulted from the closure of the

recirculation pump discharge valves. (Applicant's witness Lane, p. 6, following Tr. 15445; Staff's witnesses Hodges and Huang, p. 5, following Tr. 17943).

193 Second, the water level in the reactor vessel at TMI was not directly measured, but was inferred from water level instruments on the pressurizer. The pressurizer is a separate vessel connected to the primary system of PWR plants which normally contains both steam and water and is used to maintain system pressure such that boiling does not occur in the reactor. On BWR/6 plants such as ACNGS, the reactor vessel water level is measured directly and continuously. (Applicant's witness Lane, p. 7, following Tr. 15445; Staff's witnesses Hodges and Huang, pp. 2-3, following Tr. 17943).*/

194 Reactor vessel water level in a BWR/6 is measured by differential pressure transmitters which measure the difference in static head between two columns of water. One column is a "cold" (ambient temperature) reference leg outside the reactor vessel; the other is the reactor water in the annulus area inside the reactor

---

*/ The Board fully explored the potential causes of error in the direct measurement and determined that they are accounted for and are not of any significance. (Tr. 17970-76, 17980-84).

vessel. The measured differential pressure is a function of reactor water level. The cold reference leg is filled and maintained full of condensate by a condensing chamber at its top which continuously condenses reactor steam and drains excess condensate back to the reactor vessel through the upper level tap connection to the condensing chamber. The upper vessel level tap connection is located in the steam zone above the normal water level inside the vessel. Thus the reference leg presents a constant reference static head of water to one side of the differential pressure transmitter. The other side of the transmitter is piped to a lower-level tap on the reactor vessel. The low-pressure side of the transmitter thus senses the static head of water/steam inside the vessel above the lower vessel level instrument tap. This head varies as a function of reactor water level above the tap and is the "variable leg" in the differential pressure measured by the transmitter.

Lower taps for various instruments are located at various levels to accommodate both narrow and wide range level measurements. (Applicant's witness Lane, pp. 3-4, following Tr. 15445; Tr. 15483-97; Staff's witness Hodges and Huang, pp. 3-4, following Tr. 17943).

195 The pressure sensing devices and associated control room indicators and recorders are fully redundant as to function. There are eleven separate differential pressure sensing channels and control room indicators and recorders. Each water level range in the reactor vessel is overlapped by more than one separate sensing/indicating channel. There are two wide range level indicators/recorders and one wide range indicator, one narrow range level indicator/recorder and three narrow range indicators, one fuel zone indicator and an indicator/recorder, a high water level upset range indicator/recorder (overlaps the narrow range and wide range indicators and recorders) and a shutdown wide range level indicator (overlaps the upset range recorder). The narrow range instruments are used to indicate water level for normal plant operation and the wide range instruments are used for ECCS initiation as a result of a low water LOCA transient. (Applicant's witness Lane, pp. 4-5, following Tr. 15445; Staff's witnesses Hodges and Huang, p. 4, following Tr. 17943; Tr. 17952-55).

196 The differential pressure detectors currently used for water level measurement are recognized in the industry as among the best quality available and have a history

of being highly reliable. (Staff's witnesses Hodges and Huang, p. 6, following Tr. 17943; Tr. 17947). These sensors are rugged, contain few moving parts, are very accurate and have been fully qualified to meet applicable IEEE standards and NRC regulatory guides. The complete redundancy in function of the water level detection instruments which is provided is sufficient to accommodate any anticipated failure of the differential pressure detectors. (Applicant's witness Lane, p. 5, following 15445; Tr. 15504-05; Staff's witnesses Huang, Tr. 17956-57). The Board finds that there is no reason to require Applicant to install additional water level indicators which are redundant as to both function and type.

Doherty Contention 42: Position Indication for SRV's

197 Mr. Doherty has alleged in Contention 42 that the TMI incident has raised a concern about the potential for ambiguous indication of position of safety and relief valves.

198 According to the NRC Staff's witness, Mr. Marvin W. Hodges, TMI and most other reactors had used thermocouples to indicate flow through relief and safety

valves. At TMI-2, both thermocouples and an indication of an open or closed signal to the power operated relief valves were used. Both of these signals are potentially ambiguous: (a) the open/close signal indicates what the valve was commanded to do, not what it had done; and (b) the thermocouple can heat up due to a leaking valve or remain hot for a period of time after a valve has closed, thus providing the operator with a false position indication. (Staff's witness Hodges, p. 2, following Tr. 15128).

199 As a result of the lessons learned from TMI-2, all reactors are now required to have a direct, unambiguous indication of valve position for safety and relief valves. (Staff's witness Hodges, p. 2, following Tr. 15128). As indicated in Appendix O, page O-93 of the PSAR (App. Exh. 27) ACNGS will provide direct indication of SRV position in the main control room. This commitment was made in response to Item II.D.3 of NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License", which requires a demonstration that design and implementation can be completed prior to the issuance of an operating license. (Applicant's witness Bailey, p. 15, following Tr. 16146; Staff's witness Hodges, p. 2, following Tr. 15128).

200 SRV position indication at ACNGS will be determined by pressure measurement in the SRV discharge pipe. The pressure sensor will provide a direct indication of flow through the SRV. When the SRV is closed the pipe pressure is approximately the same as the containment pressure. As soon as the SRV opens, an instantaneous pressure rise would be detected by the sensor and would trigger the alarm. This is in contrast to the thermocouple system at TMI which could give an indication that the valve was open when, in fact, it was closed. (Staff's witness Hodges, p. 3, following Tr. 15128; Applicant's witness Bailey p. 15, following Tr. 16146).

201 The actual pressure setpoint to be used at ACNGS will be determined from a combination of analysis and field test data, and will be submitted with the Final Safety Analysis Report. Indication in the Main Control Room will be on two light matrices (one for each division of position measurement) on the Reactor Core Cooling System's benchboard above the manual control switches for the relief valves. The indication will be redundant, safety grade, seismically and environmentally qualified, and powered from a Class IE power source. (Staff's witness Hodges, p.2, following Tr. 15128).

202 This type of pressure sensing device has operated successfully in both nuclear and non-nuclear facilities for many years. (Staff's witness Hodges, p. 4, following Tr. 15128). Therefore, the Board finds that the SRV position indication design is technically feasibile, represents the state-of-the-art design and can be appropriately implemented prior to issuance of the operating license (Staff's witness Hodges, pp. 3-4, following Tr. 15128; Applicant's witness Bailey, pp. 15-16, following Tr. 16146). While Mr. Doherty raised a concern that had some basis in light of potential problems with prior designs, the Board finds that his concern has been resolved in the ACNGS design.

Doherty Contention 43: Stainless Steel Cleaning

203 In its Order of September 1, 1981, the Board granted Applicant's motion for summary disposition of Doherty Contention No. 43. However, in that Order it was noted that Mr. Doherty had raised an issue as to whether certain ECCS components and piping would be coated in accordance with the recommendations of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants."

Applicant and Staff were requested to present evidence on this single matter.

204 The Applicant has determined that areas which are not "significant" and which do not have a "direct potential post-LOCA debris pathway to the Suppression Pool" are excluded from the coating requirements of Reg. Guide 1.54 because they do not represent a potential source of debris which might clog the ECCS suction filters. "Significant areas" are those areas of the containment which would be exposed to post-LOCA containment spray and/or areas in which a LOCA may occur. "Areas which have a direct potential post-LOCA debris pathway to the Suppression Pool" are those areas of the containment which would gravity drain rapidly and with little filtering effect to the suppression pool. Thus, all major structures and equipment will be painted in accordance with the requirement of Reg. Guide 1.54. Only small areas such as valve handles will be excluded since paint that peeled off such a small area could not clog the ECCS suction filters. The Staff has determined that Applicant's position is in conformance with Reg. Guide 1.54. (Applicant's witness Malec, p. 7, following Tr. 19848; Applicant's witness Gordon, p. 4,

following Tr. 19884; Staff's witness Litton, p. 3, following Tr. 20014; Applicant's witness Malec, Tr. 19966-67). The Board finds that this information is dispositive of the remaining issue noted in the September 1 Order.

Doherty Contention 45: Core Lateral Support

205 In Contention 45, Mr. Doherty alleges that the lateral support of the ACNGS reactor core is not sufficient to withstand lateral seismic forces combined with the lateral blowdown force that arises simultaneously during a LOCA transient.*/

206 Applicant's witnesses testified that the ACNGS core structures and fuel assemblies are designed to take into account loads resulting from a seismic event in combination with loads from a loss-of-coolant accident (LOCA). With respect to core structures (shroud, core plate, control rod guide tubes, fuel supports, top

*/ In its Second Order Ruling Upon Motions For Summary Disposition (September 1, 1981), pp. 55-57, the Board denied the Staff's motion for summary disposition on this issue.

guide, shroud head and separators assembly), Applicant's witness described the impulsive loads which were assumed to apply to the core structures and which were taken into account in the stress analysis of these structures. The analysis showed that these structures will retain their integrity under these loading conditions. (Applicant's witness Hobson, pp. 3-4, following Tr. 20109). With respect to fuel assemblies, Applicant's witness testified that General Electric had evaluated the capability of BWR/6 fuel assemblies to withstand combined LOCA and seismic loads and concluded that the assemblies could withstand such loads. This analysis was documented in NEDE-21175-P, "BWR/6 Fuel assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings." (Applicant's witness Dunlap, p. 5, following Tr. 20109). The methodology in this analysis was approved by the Staff (Staff's witness Grubb, p. 2, following Tr. 20182; Tr. 20184-85). This analysis was based upon a standard assembly designed to withstand a 0.3g ground acceleration seismic event whereas the seismic requirement for ACNGS is only 0.1g. Since ACNGS will be supplied with the standard BWR/6 fuel assembly, but has a lesser seismic requirement, the GE analysis described in

NEDE-21175-P is applicable to ACNGS (Applicant's witness Hobson, Tr. 20121-22; Staff's witness Grubb, p. 4, following Tr. 20182).

207 While the analysis in NEDE-21175-P is applicable to ACNGS, that analysis did not consider all of the relevant lateral LOCA loads; in particular the asymmetric component of the LOCA load was not considered. (Staff witness Grubb, p. 4, following Tr. 20182). However, Applicant has committed to take this component into consideration in its design. (Staff's witness Grubb at 4, following Tr. 20182; See Appendix C to Supp. 2 to SER, Staff Exh. 19). In Staff's view, the increase in LOCA loads for ACNGS because of this component will be small, and will be more than offset by the lower seismic requirements relative to those in the generic report. (Staff's witness Grubb at 5, following Tr. 20182).

208 The NRC has recently published its acceptance criteria for fuel assembly structural response to externally applied forces in Revision 2 Appendix A to Section 4.2 of the Standard Review Plan (July, 1981). Although review by the Staff of GE's analysis in NEDE-21175-P took place prior to publication of Revision 2 to the SRP, Staff witness Grubb testified that the methods

used in reviewing the GE analysis were consistent with those methods outlined in SRP § 4.2, Revision 2. (Staff's witness Grubb, p. 5, following 20182). The Board gives great weight to this opinion since Mr. Grubb not only reviewed NEDE-21175-P, but authored NUREG/CR-1018 "Review of LWR Fuel System Mechanical Response With Recommendations for Component Acceptance Criteria" which provided the basis for the SRP requirements. In any event, the final calculations for ACNGS and review by the Staff are more appropriately made at the operating license stage when the fuel design is finalized. (Staff's witness Grubb, p. 6, following Tr. 20182).

209 The Board finds that Applicant has taken into account in the design of the ACNGS reactor core combined seismic and LOCA loads. The Board is satisfied that the ACNGS core will be able to withstand these loadings based upon the Staff's review of GE's generic analysis contained in NEDE-21175-P and the fact that the seismic loads for ACNGS will be less than those considered in GE's generic analysis while LOCA loads should be the same. The Staff's newly issued acceptance criteria can be appropriately applied to the ACNGS analysis at the operating license stage. (Staff's witness Grubb, p. 6, following Tr. 20182).

Doherty Contention 47: Turbine Missiles

210 Mr. Doherty contends that the ACNGS main generator turbine is not adequately designed to prevent damage to critical plant components from turbine missiles. He also contends that rapid halting of the turbine following loss of a turbine blade will damage the plant steam system. Mr. Doherty bases his contention on incidents which occurred at Yankee-Rowe and Zion nuclear plants.

211 Mr. Doherty's reliance on turbine failures at Yankee-Rowe and Zion is not meaningful. The failure at Yankee-Rowe and the cracking found in the Zion turbine were caused by stress corrosion cracking (SCC). Certain features of the GE turbine to be used at ACNGS make it less susceptible to SCC induced failures. These features are:

(1) the use of quasi-rectangular keyways, and

(2) the use of newer steels such as the Nickel-Chromium-Molybdenum-Vanadium (NiCrMoV) steel that will be used at ACNGS.

There has never been any stress corrosion cracking found in quasi-rectangular keyways such as the ACNGS design. Moreover, the use of NiCrMoV steel has substantially increased the critical crack size in turbine wheels, i.e. the size flaw that must exist in the wheel material must be larger than the critical crack size for generation of a missile to occur during operation of the turbine. In fact, the critical crack size is now substantially in excess of the size of crack that can be detected during the routine inspection program. (Applicant's witness Tees, at pp. 3-7, following Tr. 13073; Staff's witness Litton, pp. 3-5, following Tr. 13146). The possibility of turbine failure is also reduced by the fact that GE has developed an ultrasonic inspection procedure for the wheels without disassembly of the turbine. (Staff's witness Litton, p. 5, following Tr. 13146).

212 Notwithstanding the low probability of a turbine failure, Applicant has calculated the probability of a turbine missile impacting safety related structures by evaluating and combining three conditional probabilities: the probability of a turbine failure with missile ejection, P1; the probability of a damaging strike on a structure, P2; and the probability of damage to the structure or components housed there, P3.*/ The result of

---

*/ Dr. Iotti, Applicant's exp t witness on the probability calculation, relied upon a compilation of the historical data of turbine failures in calculating the probability of missile generation. This is a very conservative calculation because: (1) there has never been a failure of a GE nuclear turbine, and (2) the ACNGS turbine design is a significant improvement over the older designs included in the data base. (Tr. 13083-4; and see Tr. 13129-31).

this calculation shows that the risk of damage is below $10^{-7}$ per year, which the Board finds is of sufficiently low probability to preclude the need to be considered in the design of the plant. (Applicant's witness Iotti, at pp. 2-4, following Tr. 13073; Staff witness Litton, pp. 2-3 following Tr. 13146; Tr. 13074-76). Moreover, the Board finds that rapid stoppage of a turbine due to missile ejection is not a safety concern because the turbine and main steam system downstream of the containment building are not safety related systems. Halting the turbine would be no worse than rapid closure of the main steam isolation valves which the plant is designed to accommodate. (Applicant's witness Carnes, pp. 4-5, following Tr. 13073; Tr. 13102-3; 13137-42).

Doherty Contention 48: CRD Return Line.

213 In his Contention 48, Mr. Doherty alleges that "ACNGS should be designed with a control rod drive return line, because this source of high pressure water functions as an additional safeguard against events where there is water loss from the reactor vessel yet pressure remains high." In support of his contention, Mr. Doherty cites incidents at Browns Ferry, Dresden II,

and Oyster Creek in which the Control Rod Drive (CRD) system, in its normal mode of operation, provided a small amount of reactor coolant makeup water by pumping water through the CRD return line and the control rod drives.

214 The CRD system provides hydraulic operating pressure and cooling water for the control rod drive mechanisms. (Applicant's witness Ross, pp. 1-3, following Tr. 12690). The original ACNGS design did include a CRD return line which was designed to return water in excess of the CRD system requirements to the reactor vessel. The ACNGS pressure vessel, which has already been fabricated, includes a CRD return line nozzle. However, the CRD return line was eliminated from the ACNGS design when cracks were found in the return line nozzle and the surrounding reactor vessel wall of other reactors. (Applicant's witness Ross, p. 4, following Tr. 12690).

215 Studies of crack initiation in such nozzles indicated that the cracks were attributable to thermal fatigue (Applicant's witness Ranganath, Tr. 13282). As a result of such studies, General Electric made a series of recommendations, including the cutting and capping of the line and nozzle on existing plants and the

elimination of the line completely on new plants. (Applicant's witness Ross, pp. 3-4, following Tr. 12690; Staff's witness Leung, p. 4, following Tr. 12992). Applicant's witness Dr. Ranganath testified that the elimination of the CRD return line would prevent such crack initiation (Tr. 13238). In NUREG-0619, the Staff concurred with the recommendation that the CRD return line be eliminated from the BWR/6-238 design used at ACNGS. (Applicant's witness Ross, p. 4, following Tr. 12690; Staff's witness Leung, pp. 4-5, following Tr. 12992; Tr. 13015-16).

216 The removal of the CRD return line and nozzle, however, would not eliminate the CRD system as a source of reactor vessel makeup water. Exhaust water from a moving CRD is disbursed to the non-moving drives via a reverse flow from the CRD hydraulic system exhaust header through the insert exhaust solenoid valves on the HCU of the non-moving CRD. A small exhaust water header back pressure resulting from drive movement lifts the solenoid valve pistons of the adjacent non-moving CRD and permits exhaust flow to be discharged through the non-moving CRD seal mechanisms and into the reactor. CRD mechanisms are designed to function with small bypass flow to the reactor vessel. Thus,

the CRD system without the return line does have the capability of providing an additional source of reactor coolant makeup water. (Applicant's witness Ross, pp. 4-5, following Tr. 12690).

217 In doing its safety analyses, General Electric does not take any credit for the CRD system's capacity to provide makeup water. Nonetheless, GE performed an analytical comparison of CRD system injection capability for various BWR designs before and after deletion of the CRD return line, using as a model the incident at Browns Ferry I which, of the three incidents cited by Mr. Doherty, was the one that placed the greatest demands on the CRD system. Under those conditions, the flow necessary to keep the core from uncovering was calculated and compared for CRD systems with and without the CRD return line and with one or two pump operation. The calculations were weighted to maximize the amount of water needed to keep the core covered, and to emphasize any apparent change due to elimination of the return line. The results of GE's comparison for a BWR/6-238, such as ACNGS, show that the injection flow rate for the CRD system provides enough makeup flow to keep the core covered under the conditions of the Browns Ferry I fire. (Applicant's witness Ross, pp. 6-7, following Tr. 12690).

218 The Board finds that there is no reason to require Applicant to redesign ACNGS to include a CRD return line. The return line was eliminated to resolve a cracking problem; it is not needed for proper operation of the CRD system and the CRD system without the CRD return line still provides a coolant injection path to the reactor. Accordingly, the Board finds no merit to Doherty Contention 48.

Doherty Contention 50: Jet Pump Beam

219 In his Contention 50, Mr. Doherty alleges that the beam which holds the jet pumps in place in the ACNGS reactor system will crack and thereby subject the coolant recirculation system to disassembly and/or hazardous displacement. The basis of Mr. Doherty's concern is an incident at Dresden Unit 3 where a jet pump failed.

220 During normal reactor operation, the function of the jet pump is to develop the required driving head to the primary coolant for the purpose of circulating the coolant through the reactor core. A secondary function of the jet pump is to provide core flow measurements through calibrated diffusers. There are a total of 20 jet pumps located in the annular region between the

core shroud and the vessel inner wall. The jet pumps also have a safety function during the design basis Loss-of-Coolant-Accident (LOCA). For the LOCA event, the jet pumps allow post-accident flooding of the core to no less than two-thirds of the core height. This feature assures that a sufficient level of water will remain in the core such that along with Emergency Core Cooling Systems the core will remain within 10 CFR 50, Appendix K limits. (Applicant's witness Aleksey, p. 1, following Tr. 13236).

221 The jet pump failure at Dresden 3, cited by Mr. Doherty, was attributed to the failure of the Inconel X-750 jet pump hold down beam. Subsequent examination of other BWR facilities revealed cracked hold down beams in nine other plants. (Staff's witness Litton, p. 2, following Tr. 13183).

222 Of the BWR's that have experienced cracks in the jet pump beams, all have been BWR-3 designs (Staff's witness Litton, p. 2, following Tr. 13183). For the BWR-6 designs, such as ACNGS, the potential for cracking has been lowered by the use of thicker hold down beams. (Staff's witness Litton, p. 4, following Tr. 13183; Applicant's witness Aleksey, p. 4, following Tr. 13236).

223 In addition to being thicker, the hold down beams to be used at ACNGS will include additional design features which should further reduce the risk of cracking. The cause of crack initiation and failure of all beams evaluated to date has been Intergranular Stress Corrosion Cracking (IGSCC). IGSCC is the cracking of metals along their grain boundaries and is caused by a combination of relatively high stress, a corrosive environment and a physical characteristic of the metal which renders grain boundary regions of the material susceptible to local corrosive attack. (Applicant's witness Aleksey, p. 3, following Tr. 13236; Staff's witness Litton, p. 2, following Tr. 13183).

224 General Electric has determined that by changing the heat treatment of the metal, Inconel X-750, and reducing the stresses imposed on the beam, the potential for jet pump beam cracking is substantially reduced. Heat treating of metal is done to ensure that the desired grain structure and grain boundary composition is formed in the metal. The original heat treatment, used on the jet pump beams which experienced cracking, was to heat the material to 1625°F and hold it there for 24 hours, then subsequently heat it to 1300°F and hold it there for 20 hours. The heat treatment to be used on

the newer beams will be to heat the material to approximately 2000°F for 1 hour and subsequently heat it to 1300°F and hold it there for 20 hours. This new heat treatment will strengthen the grain boundaries and lessen susceptibility to IGSCC. This can be done because the new heat treatment, high temperature annealing, softens the metal and produces a recrystallized grain structure. This grain structure is less susceptible to IGSCC and when it is then age hardened at 1300°F for 20 hours, the metal attains its final required hardness (strength) with the improved grain structure. (Applicant's witness Aleksey, pp. 4-5, following Tr. 13236; Staff's witness Litton, p. 3, following Tr. 13183).

225 In addition, in response to the cracking problem, General Electric has further reduced the stresses in the beam by reducing the beam preload from 30 kips to 25 kips. (Staff's witness Litton, p. 3, following Tr. 13183; Tr. 13247).

226 The Applicant has agreed to purchase the new heat treated beams (Applicant's witness Root, Tr. 13268, 13270; App. Exhs. 24, 24-A and 24-B). The Board finds that the combination of (a) the heavier cross section, (b) the high temperature annealing treatment and (c)

the lowered preloading stress should enable the jet pump hold down beams at ACNGS to have a service life of more than 40 years (Tr. 13262; Staff's witness Litton, p. 3, following Tr. 13183) and reduce the likelihood of IGSCC which has been identified as the cause of cracking in the hold down beams. (Applicant's witness Aleksey, p. 6, following Tr. 13236). Accordingly, the Board finds no merit in Mr. Doherty's contention.

McCorkle Contention 9: Chlorine Monitoring

227 In her Contention 9, Intervenor McCorkle alleges that the control room operators are inadequately protected against the hazards of chlorine or other toxic gases. She alleges that in the absence of such protection, evacuation of the control room may be required before the plant could be brought to safe shutdown.

228 The only material to be stored on-site that could pose a potential hazard to control room personnel is chlorine. (Applicant's witness Martin, p. 2, following Tr. 16245). In PSAR Appendix C (App.Exh. 27), the Applicant has committed to provide protection for control room personnel from potential accidents involving the release of chlorine gas. The design of the control room, therefore,

includes detectors located in the fresh air inlets which, upon detection of chlorine, will provide an audible alarm and automatically isolate the control room. These detectors will also detect chlorine releases which may originate from off-site sources, including the nearby Atchison, Topeka and Santa Fe (AT&SF) Railroad. In addition, adequate self-contained breathing apparatus will be provided for further protection of control room operators. (Applicant's witness Martin, pp. 2-3, following Tr. 16245).

229 The Staff has reviewed the Applicant's proposed Control Room Habitability System design with respect to protection against chlorine and has found that it is consistent with NRC Standard Review Plan 6.4 (NUREG-75/087) and meets the guidelines of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release". (Staff's witness Campe, pp. 5-6, following Tr. 17314).

230 An accident involving a train traveling along the AT&SF Railroad line which is located approximately 4500 feet west of the control room, presents the only possible off-site source of toxic gases, other than chlorine, of sufficient quantity to pose a threat to control room operators. (Applicant's witness Martin, p. 3, follow-

ing Tr. 16245). Both the Staff and the Applicant continue to gather information as to the nature and frequency of shipments of hazardous chemicals past the ACNGS site as well as the nature and frequency of accidents on such line. (Applicant's witness Martin, p. 4, following Tr. 16245; Staff's witness Campe, p. 6, following Tr. 17314). The results of such data gathering activities will be available at the operating license stage for ACNGS.

231 Since the capability for control room isolation is already a part of the control room design, in the event that such data discloses a probability of additional toxic gas hazards that exceed the guidelines of Regulatory Guide 1.78, appropriate steps can be taken at that time to adequately protect control room personnel. (Staff's witness Campe, p. 7, following Tr. 17314). There are detectors currently available which can detect all types of toxic gases and the installation of such detectors or appropriate warning systems could be accomplished at ACNGS with only minor alterations to the control room. (Applicant's witness Martin, p. 5, following Tr. 16245).

232 The Board finds that the ACNGS control room is adequately protected against the hazards of chlorine gas

and that appropriate measures can be taken at the operating license stage to protect it, if necessary, against the hazards of other toxic gases.

McCorkle Contention 14: Fuel Rod Hydriding

233 Intervenor McCorkle contends that the ACNGS fuel rods are subject to hydride induced clad failure. She also contends that the fuel rods are subject to fuel densification which would cause increases in power spikes and the heat generation rate. On August 4, 1980, Applicant filed a motion for summary disposition of the contention. Ms. McCorkle did not file an answer. In its motion the Applicant asserted the following facts:

1. Fuel hydriding is caused by moisture or other hydrogenous materials left inside the Zircaloy fuel rod during manufacture.

2. In order to prevent hydrogen contamination of the inside of the fuel rod, two major changes have been made in the manufacturing process during or since the early 1970's. These two changes consisted of installing a hot vacuum outgassing system to remove moisture from the fuel just prior to welding the end plug of the rod in place, and of installing a hydrogen getter in the form of zirconium alloy chips inside the fuel rod to preferentially combine with hydrogen present in the rod.

3. No hydride induced failures have occurred in General Electric BWR fuel manufactured using the hydrogen getter and the outgassing techniques.

4. Knowledge of the causes of in-reactor fuel failures has led to quality control tests during manufacture which assure that the fuel is of such an initial density that further densification during irradiation does not affect the thermal-mechanical performance of the fuel. Further, conservative limits on the Linear Heat Generation Rate (LHGR) allowed in the reactor fuel assure that the actual LHGR will remain within design limits if maximum theoretically possible densification occurs.

5. No fuel cladding failures or collapses attributable to densification have ever occurred in BWR fuel.

234 In the Order of September 1, 1981, as clarified on October 7, 1981 (Tr. 18024), the Board accepted the facts in paragraphs 1, 2 and 4 as true but required Applicant to provide an update as to paragraphs 3 and 5. The Applicant called Mr. Noel C. Shirley as a witness to update his prior affidavit on this issue.*/ Mr. Shirley is a Senior Licensing Engineer at GE and is responsible for all generic licensing issues affecting BWR Fuel. (Tr. 20235). He confirmed that as of the date of his testimony the statements in paragraphs 3 and 5 were still true. (Applicant's witness Shirley, following Tr. 20218; Tr. 20226-27).

---

*/ Ms. McCorkle previously indicated she would withdraw the contention if she did not retain a witness. (See Order of January 8, 1980). She neither presented a witness nor appeared at the hearing to cross-examine on this issue.

235 Based on the foregoing uncontroverted evidence, the Board finds that the Applicant has demonstrated that the problems of fuel hydriding and densification are no longer problems in GE BWRs. These problems have been eliminated by changes in the design and manufacturing process of fuel rods. Accordingly, we conclude that the ACNGS fuel rods will not be subject to hydride induced clad failure or to fuel densification.

McCorkle Contention 17: Filtration System Leakage

236 Intervenor McCorkle alleges that the containment is designed such that 20% of the leakage from the containment will bypass the filtration system.*/

237 The ACNGS containment consists of a free-standing steel shell 1-1/2 to 1-3/4 inches thick which encloses the reactor pressure vessel. The containment is designed to protect the public from the release of radioactive fission products by providing a leak-tight barrier. How-

*/ Applicant filed a motion for summary disposition on this contention. Although Ms. McCorkle did not oppose the motion, the Board denied the motion for the reasons stated in the Order of September 1, 1981. Ms. McCorkle had previously indicated she would withdraw her contention if she did not find an expert witness. (See Order of January 8, 1980). However, she neither presented a witness nor appeared at the hearing on this issue.

ever, for practical purposes, the containment must be penetrated by piping and other openings. (Applicant's witnesses Martin and Malec, Attachment GM-1, p. 2, following Tr. 19467). Although these penetrations are sealed by means such as redundant valving, some leakage is inevitable. NRC regulations limit the quantity of leakage allowed. (See 10 CFR Part 50, Appendix J) (Staff's witness Fields, pp. 1-2, following Tr. 19497).

238 In the calculation of the off-site radiological doses to show compliance with the siting criteria of 10 CFR Part 100, the containment is assumed to leak at a constant leak rate of 0.5% of its volume per day. From a dose calculation standpoint, the radionuclides, uniformly mixed in the containment atmosphere, are assumed to leave the containment at this constant leakage rate regardless of the flow rate of carrier air in which they are assumed to be mixed. The maximum containment airborne concentration of these radionuclides will occur at standard temperature and pressure (STP) conditions. Therefore, the air leakage expressed in terms of a fraction of the containment air volume at STP conditions will have the same radionuclide concentration and hence will be selected as the technical specification value to be met, in testing, in order to remain within the dose criteria of 10 CFR

Part 100. In fact, the amount of containment leakage allowed in the technical specifications will be significantly less than that which would produce total off-site doses equal to the 10 CFR 100 limits. The actual value of the bypass leakage technical specification will be determined as a result of LOCA dose calculations performed when the Final Safety Analysis Report is prepared for submittal. Applicant has identified all of the potential pathways for unfiltered leakage and has estimated that only 0.0195 percent per day of the containment volume would bypass the filtration system under LOCA conditions. This amounts to 4% of the 0.5% allowable containment leakage, not 20%, as alleged by Ms. McCorkle. (Applicant's witness Martin, pp. 2-3 and Attachment GM-1, pp. 2-4, following Tr. 19467; Tr. 19483-85). Staff also agrees that the amount of unfiltered leakage assumed for ACNGS will be only approximately 4% of the total leakage allowed. (Staff's witness Fields, p. 2, following Tr. 19497).

239 In order to assure that the containment will maintain its expected level of leak-tightness, Applicant will conduct a leak testing program in accordance with Appendix J of 10 CFR 50. Appendix J requires extensive preoperational leak tests and periodic leak tests during the life of the

plant to assure that the containment leak rates do not exceed predicted limits. Type A leak tests (total containment leakage) will be performed three times during each 10-year service period while Types B and C leak tests (for containment penetrations), which will provide a measure of expected unfiltered leakage, will be performed at intervals not to exceed 2 years of duration. (Staff's witness Fields, p. 2, following Tr. 19497; Tr. 19499-500; Applicant's witnesses Martin and Malec, Attachment GM-1, pp. 5-6, following Tr. 19467; Tr. 19487-93).

240 The Board finds that a maximum of 0.0195 percent per day of containment volume may escape via the potential bypass leakage lines and that the resulting doses will not exceed the limits of 10 CFR Part 100. Hence, Intervenor's claim that 20% of the containment leakage will bypass the filtration system is without merit.

Scheussler Contention 1: Emergency Planning

241 In its order of March 10, 1980, the Board deferred ruling upon contentions dealing with the adequacy of Applicant's emergency evacuation plans, because the Commission had published a proposed amendment to its emergency planning rules. In an order dated July 24, 1980, the Board noted

that the proposed rule had not become final, and that in order to avoid delay in the hearings, previously filed contentions on emergency planning would be admitted. The Board admitted TexPirg Additional Contention 16(d), (e), (f) and (i) and Additional Contention 42, Doggett Contention 5, Scheussler Contentions 6 and 14. By Order dated September 19, 1980, these contentions were consolidated into Scheussler Contention 1, which reads as follows:

> I. ACNGS fails to adequately meet requirements of 10 C.F.R. Part 100 regarding siting, for reasons which include, but are not limited to, the following: (a) Applicant fails to adequately recognize that metropolitan Houston is the fastest-growing area in the U.S., steadily and rapidly expanding toward the site of ACNGS; (b) The proposed site of ACNGS is not presently sufficiently remote, and will become even less so during its operating life; (c) Traffic congestion at present and for the foreseeable future prevents any effective, timely emergency evacuation of the greater Houston area, or any substantial part thereof; (d) The distance from ACNGS to population center should be much greater than 1-1/3 X LPZ because of special circumstances cited above.
>
> II. The PSAR fails to meet requirements of 10 C.F.R. Part 50, Appendix E, II, in that it fails to assure the compatibility of emergency plans with site location, access routes, population distribution and land use.
>
> III. The PSAR and the selection of the proposed site do not properly consider population density, land use, physical characteristics thereby failing to adequately insure low risk of public exposure as required by 10 C.F.R. Part 100.10.

242 Those parts of the contention related to the suitability of the ACNGS site under Part 100 based upon population density and growth and the radiological risk to the public, namely, Parts I(a), (b) and III, are essentially disposed of in the findings on Bishop Contention 1 and TexPirg Contention 1. All that will be addressed here is the adequacy of ACNGS emergency planning as it relates to Parts I(c) and II of the contention, and Part I(d) relating to the population center distance under Part 100.

243 Subsequent to admission of the contention, the Commission published a final rule on emergency planning. (See, 45 Fed. Reg. 55402). This rule is found in 10 CFR Part 50, Appendix E.*/ Part II of Appendix E describes the information which must be contained in the PSAR at the construction permit stage. Applicant has submitted the information required by Appendix E in Section 13.3 of the PSAR (App. Exh. 27). The Staff has reviewed the PSAR submittal and has determined that Applicant complies

---

*/ Although the work underlying the new rule began prior to the accident at Three Mile Island ("TMI"), the new rule incorporates the lessons learned at TMI with respect to emergency planning. (Tr. 18665-69).

with Appendix E.*/ (SER Supp. 4, Staff Exh. 21, pp. 13-1 to 13-14; Staff's witness Kantor, pp. 20-21, following Tr. 18623).

244 Part II of Appendix E establishes only minimum requirements for an emergency plan at the CP stage. Part III of Appendix E requires that the detailed emergency plan containing all the information required by Part IV of Appendix E be submitted in the Final Safety Analysis Report. Appendix E requires detailed emergency planning, including evacuation planning, for an area of about 10 miles around the plant. This area is known as the Emergency Planning Zone (EPZ). Part II of Appendix E also requires that the PSAR contain sufficient information "to ensure the compatibility of proposed emergency plans . . . with respect to such considerations as access routes, surrounding population distributions,

---

*/ In accordance with the NRC/FEMA Memorandum of Understanding, the Federal Emergency Management Agency (FEMA) reviewed the status of State and local plans and preparedness for ACNGS. FEMA indicated that although the State of Texas has not yet prepared any site specific plans for ACNGS, it anticipates no major impediments or constraints that would inhibit such plans. Therefore, FEMA concluded that there is reasonable assurance that there are no unusual problems that can not be adequately handled as the site specific emergency plans are developed. (SER Supp. 4, Staff Exh. 21, p. E-1, E-2).

land use, and local jurisdictional boundaries . . . ." Contrary to Part II of the contention, such considerations were evaluated by both Applicant and Staff and were found to be compatible with the development of detailed emergency plans for ACNCS. (Staff's witness Kantor, pp. 3-5, 20, Tr. 18623; Staff Exh. 21, SER Supp. 4, Sec. 13).

245 With respect to access routes, the 10-mile EPZ has many well-paved highways leading in all directions which provide access routes to the plant as well as evacuation routes for the public. Major evacuation routes in the 10-mile EPZ are State Highway 36 running north and south from the plant, Farm-to-Market Road (FM) 1093 running east and west from Wallis and Interstate 10 running east and west from Sealy. (Applicant's witness Murri, pp. 5-6, following Tr. 17433). Applicant's evacuation time study, discussed below, shows that there are many other well-paved roads which provide varied evacuation routes within the 10-mile EPZ.

246 With respect to surrounding population distributions and land use, the majority of the 10-mile EPZ is sparsely-populated rural farming land. There are no major metropolitan areas within or near the 10-mile EPZ. Only the small towns of Wallis and Sealy and the commu-

nities of Orchard and Simonton are located within the 10-mile EPZ. Specific land use activities within the 10-mile EPZ include the ACNGS Lake and State Park, the Wallis School, the Sealy School, the Orchard School, the Brazos High School, the Stephen F. Austin State Park, the Sealy Medical Center and the Azalea Manor Nursing Home. Both the Medical Center and Nursing Home are located in the town of Sealy. Each of these schools, parks and institutions are located along major highway routes and do not present any obstacle to the development of detailed emergency plans. (Applicant's witness Murri, p. 6, following Tr. 17433; Staff's witness Kantor, p. 18, following Tr. 18623; Staff Exh. 21, SER Supp. 4, pp. 13-11, E-1 and E-2).

247 Applicant has also taken into consideration local jurisdictional boundaries in the emergency planning information submitted in the PSAR. Parts of five counties are included in the 10-mile EPZ. HL&P has consulted with the County Sheriff's Department in each of these five counties. The County Sheriff's Department is, in effect, the local agency with responsibility for implementation of emergency actions within their respective counties. (Staff's witness Kantor, p. 11, following Tr. 18623; Tr. 18792). HL&P has also consulted with

the Texas Department of Health, which is the State of Texas agency with lead responsibility for the development of State emergency plans. (Staff's witness Kantor, pp. 11, 16, following Tr. 18623; Applicant's witnesses Murri and Lawhn, Tr. 17528-32; 17556-59). Each of these agencies has agreed to cooperate with HL&P in the development of the detailed emergency plan for ACNGS. (Applicant's witness Lawhn, p. 7, following Tr. 17433; Staff's witness Kantor, pp. 5-9, following Tr. 18623). It is also noteworthy that there are two other nuclear power plant projects that will be in operation in Texas prior to operation of ACNGS. Thus, the State's emergency plan for nuclear plants will be finalized long before ACNGS goes into operation. (Applicant's witness Lawhn, Tr. 17504-5).

248 As part of the process of evaluating the feasibility of developing a detailed emergency plan, Applicant hired Mr. Bill Griffin of HMM Associates to study the estimated time required to evacuate various sectors and distances within the plume exposure pathway EPZ for both transient and permanent populations, noting major impediments to evacuation or taking of protective actions. Mr. Griffin and his firm have done comparable studies for several other nuclear plants. (Applicant's witness Griffin,

Tr. 17436-39). The study was done using a computer program which simulates evacuation by vehicle. This computer program provides a method to estimate evacuation times and to identify any major impediments to evacuation or the taking of protective actions. The results of the evacuation time calculations are summarized in Table 5-1 of the study report, which is Appendix 13.3B of the PSAR (App. Exh. 27). The estimates in the table represent the elapsed time from notification of the public to the time when all people have left the area. This analysis compared evacuations during good and bad weather. It also compared evacuations during permanent and peak population conditions. The study showed that the 1990 population of the entire EPZ could be evacuated in under two hours, even during adverse weather, and that there were no major impediments to evacuation. Indeed, Mr. Griffin testified that the estimated evacuation time for ACNGS is one of the quickest of the sites that had been evaluated by his firm. (Applicant's witness Griffin, pp. 8-9, following Tr. 17433; Tr. 17444-45).

249 In this same regard, Staff's witness Mr. Urbanik independently confirmed that there were no impediments to evacuation of the site and that the evacuation time

estimates were reasonable, regardless of the adequacy of the emergency plan. Mr. Urbanik's review of the evacuation time studies considered such factors as (a) an accounting for permanent, transient, and special facility populations in the plume exposure EPZ; (b) an indication of the traffic analysis method and the method of arriving at road capacities; (c) a consideration of a range of evacuation scenarios generally representative of normal through adverse evacuation conditions; (d) consideration of confirmation of evacuation; (e) identification of critical links and need for traffic control; and (f) use of methodology and traffic flow modeling techniques for various time estimates. (Staff's witness Urbanik, pp. 4-5, following Tr. 18632; Tr. 18754). Mr. Urbanik is a transportation specialist who is working under an independent contract with the NRC. He was the principal author of NUREG/CR-1745, "Analysis of Techniques for Estimating Evacuation Times for Emergency Planning Zones," and he assisted in development of the guidance for evacuation time estimate studies contained in Appendix 4 to NUREG-0654. While under contract with the NRC he has reviewed the evacution time estimates for every nuclear plant in the United States. (Staff's witness Urbanik, Tr. 18826;

18863). Both Mr. Urbanik and Mr. Griffin testified that there is absolutely no basis for Part I(c) of the contention, because traffic congestion in Houston will not interfere in any way with evacuation of the 10-mile EPZ. (Applicant's witness Griffin, Tr. 17443-44; 17514-16).*/

250 The conclusions drawn by Mr. Griffin and Mr. Urbanik were underscored by the testimony of Mr. Murri, who had been retained by Applicant to assist in development of emergency plans for both ACNGS and the South Texas Project nuclear plant. Mr. Murri has extensive experience in the field of emergency planning and that he is in fact one of the pioneers in the field of developing emergency plans for nuclear plants. (Applicant's witness Murri, Tr. 17419). Mr. Murri testified that he saw no potential obstacles which would preclude the development of the detailed emergency plan required at the operating license stage. He testified that in his judgment the relatively low population density and other favorable

---

*/ To the extent that the contention may be read as alleging that Applicant should develop a plan for evacuating Houston, the contention constitutes an impermissable challenge to the Commission's regulations, specifically 10 CFR Part 50, Appendix E.

site area characteristics within the 10-mile EPZ make the ACNGS site compare favorably for emergency planning to the numerous other nuclear plant sites with which he was familiar. (Applicant's witness Murri, p. 9, following Tr. 17433).

251 In addition to meeting the specific allegation in Scheussler Contention 1, Staff's witness Falk Kantor described each individual requirement in Part II of Appendix E, and the steps the Applicant has taken to comply with those requirements. Mr. Kantor also has extensive experience in the field of emergency planning for nuclear power plants. Mr. Kantor is presently employed by the NRC as an Emergency Planning Analyst in the Emergency Preparedness Licensing Branch in the Division of Emergency Preparedness. In this capacity he is responsible for the review and evaluation of radiological emergency plans submitted by reactor applicants and licensees. Mr. Kantor has reviewed the Applicant's emergency plan and has determined that it fully complies with the applicable provisions of Appendix E, Part II. (Staff's witness Kantor, pp. 5-20, following Tr. 18623). Based on its review of Items A through M as described in Mr. Kantor's testimony and in Section 13 to the SER Supp. 4 (Staff Exh. 21), the Staff has concluded that the Allens

Creek PSAR contains sufficient information to ensure that the proposed emergency plans for both onsite areas and the EPZs are compatible with facility design features, site layout, and site location with respect to access routes, surrounding population distributions, land use, and local jurisdictional boundaries. (Staff's witness Kantor, p. 20, following Tr. 18623; Staff Exh. 21, SER Supp. 4, p. 13-13). In addition, 10 CFR § 50.47(b) lists 16 planning standards that must be met in the emergency response planning for a nuclear power reactor. The Staff has reviewed the Applicant's response to these standards and has concluded that the information presented is sufficient in depth and scope for the CP stage to indicate the feasibility of meeting the planned standards in the final Emergency Plan. Further, no special or unique circumstances have been identified which would preclude the development of adequate emergency preparedness plans at the OL stage of review. Based on its assessment of the Applicant's emergency plans and the FEMA findings regarding the status of offsite emergency planning, the Staff has concluded that the overall state of emergency preparedness for ACNGS is acceptable for the CP stage of review. (Staff's witness Kantor, p. 21 following Tr. 18623; Staff Exh. 21, SER Supp. 4, pp. 13-13, 13-14).

252 The Board concurs with this conclusion, and finds that in demonstrating compliance with Appendix E, Part II, those portions of Schuessler Contention 1 specifically addressed to the adequacy of the emergency plan for ACNGS have been considered and assessed. Accordingly, we conclude that Parts I(c) and II of Schussler Contention 1 have no merit.

253 With respect to Part I(d) of the contention, Mr. Scheussler seems to argue that because of the potential difficulty in evacuating Houston, it should be the "nearest population center." Mr. Schuessler apparently has a misunderstanding about the relationship between Part 100 siting criteria and emergency planning. The NRC regulations require that an emergency plan be developed for a 10-mile EPZ surrounding the proposed plant regardless of where the population center under Part 100 might be located. Therefore, emergency planning considerations do not provide any "special" reason for moving the population center. As a result of the new regulations in Appendix E, Applicant is now required to plan for evacuation of a much larger area than the low population zone (LPZ).*/

*/ Under prior regulations, an applicant was required to establish evacuation plans for people located within the LPZ. See 10 CFR § 100.3(b). The size of the LPZ was determined by a maximum radiological dose assumed to be received by an individual from a postulated accident at the plant. See 10 CFR § 100.11(a)(2).

Accordingly, the concepts of LPZ and population center have no real bearing on emergency planning. It is noteworthy that Applicant did initially pick Houston as the population center but at the Staff's suggestion it later selected the Richmond/ Rosenberg area as the population center in order to be more conservative. (See SER, Staff Exh. 23, p. 2-4). This area is approximately 20 miles from the ACNGS site and it is still more than 4 times the minimum distance permitted in Part 100. Moving the population center further away from the site (i.e., to Houston) is less conservative. (Applicant's witness Lawhn, pp. 9-10, following Tr. 17433).*/ Thus, the Board finds that the designation of the nearest population center has been in accordance with NRC regulations and that there is no reason to designate Houston or any other city as the nearest population center.

---

*/ As discussed in the findings on Bishop Contention 1, the Staff also considered the towns of Sealy and Katy as the nearest population center, however, it could not designate either of these towns as the nearest population centers because of the uncertainty of long range population projections. The Staff noted, however, that if either Sealy or Katy should become the nearest population center, the distance from the plant site to either of these cities would still be greater than 1-1/3 times the low population zone outer radius of 3.5 miles. (Staff Exh. 19, SER Supp. 2, p. 2-4).

TexPirg Contention 6 (McCorkle Contention XI): Aircraft Hazards.

254 TexPirg's Contention 6 alleges that the Applicant has understated the probability of a large aircraft, such as a Boeing 747, crashing into the ACNGS containment vessel. As a result, TexPirg alleges that the Applicant has failed to consider the maximum credible accident that could occur. The bases for TexPirg's contention are its allegations (1) that large aircraft traffic in the Houston area has increased in the past few years and will continue to increase throughout the life of the plant and (2) that new airports capable of handling such large aircraft have been proposed in the area of the plant.

255 Aircraft accidents may be associated with airport operations, i.e., takeoffs and landings, or with "in-flight" operations. With respect to airport operations, the accident rate decreases as distance from an airport increases. (Applicant's witness Woodard, p. 3, following Tr. 11316). Accordingly, the NRC in its Standard Review Plan (NUREG-75/087), Section 3.5.1.6, requires a statistical analysis of the probability of an aircraft crash only if the number of airport operations and the distance

from the plant to the airport fail to satisfy certain criteria.*/ There are currently no airports with commercial traffic within ten miles of the ACNGS site nor is there any evidence that any such airport is proposed. The only airports within fifty miles of the ACNGS site which are capable of handling large commercial aircraft are Houston's Intercontinental and Hobby airports, both of which are more than 45 miles from the ACNGS site. Of these, only Intercontinental is capable of handling heavy aircraft such as the Boeing 747. (Applicant's witness Woodard, pp. 2-3, following Tr. 11316).

256 Under the NRC's Standard Review Plan, a statistical analysis of accident probabilities at either Intercontinental or Hobby would be necessary only if the number of operations at such airports exceeded 2,225,000 annually. While there have been recent increases in traffic

---

*/ Section 3.5.1.6 of the Standard Review Plan requires a statistical analysis for the following:

a. Any airport located within 5 miles of the site;

b. An airport with projected operations greater than 500 $d^2$ movements per year located within 10 miles of the site;

c. An airport with projected operations greater than 1000 $d^2$ movements per year located beyond 10 miles from the site, where "d" is the distance in miles from the site.

in the Houston area there are presently only about 335,000 annual operations at Intercontinental and about 350,000 annual operations at Hobby. The Federal Aviation Administration projects that in 1992 annual operations at Intercontinental and Hobby will increase, but only to about 526,000 and 494,000, respectively. (Applicant's witness Woodard, p. 4, following Tr. 11316; Staff's witness Campe, p. 2, following Tr. 11360).

257 When measured by the criteria of Section 3.5.1.6 of the Standard Review Plan, neither the current nor the projected number of operations at Intercontinental or Hobby require a statistical analysis of accident probabilities. Moreover, the witnesses for both the Applicant and the Staff testified that at distances as great as those involved at Hobby and Intercontinental, the probabilities of a crash would be so low as to be subsumed by the statistics on in-flight operations (Staff's witness Campe, Tr. 11376; Applicant's witness Woodard, Tr. 11329). In fact, the Staff testified that it is generally not concerned with activities at airports beyond 20 miles and, therefore, it is extremely conservative to use the NRC criteria in this instance where the airports are 45 miles from the site. (Tr. 11376).

258 Using an equation set forth in the Standard Review Plan (Section 3.5.1.6), Applicant's witness Woodard made an analysis of the probability of an in-flight accident involving large aircraft affecting the plant and determined that the probability of such an event is $1 \times 10^{-8}$ per year for large commercial aircraft (jet aircraft such as the Boeing 707, 727) and $2.1 \times 10^{-10}$ per year for heavy aircraft (large jet aircraft such as the Boeing 747, DC 10). Doubling these values to account for the increased commercial aircraft traffic expected by 1992 yields probabilities of $2 \times 10^{-8}$ per year for large commercial aircraft and $4.2 \times 10^{-10}$ for heavy aircraft.

259 The Board finds that these probabilities are well below the NRC acceptance criteria and consequently neither design changes nor further inquiry is necessary.

260 In addition, the record contains no evidence supporting TexPirg's allegation that new airports might increase

the hazards posed by aircraft to ACNGS. Both Mr. Woodard and Mr. Campe acknowledged that applications to build "reliever" airports at various sites ranging from 15-20 miles east and northeast of the site had been received by the FAA. (Applicant's witness Woodard, p. 9, following Tr. 11316; Staff's witness Campe, p. 3, following Tr. 11360). However, none of these applications has received FAA approval and all but one has now expired. The remaining application contains a proposal to locate an airport some nineteen miles from the site. The FAA estimates that such an airport could have 100,000 to 200,000 annual operations, most of which would be smaller general aviation aircraft. The Board finds that this facility, even if built, is too distant from the plant to generate sufficient operations to require either a further statistical analysis or changes in plant design. (Applicant's witness Woodard, pp. 8-9, following Tr. 11316; Staff's witness Campe, p. 3, following Tr. 11360).

TexPirg Additional Contention 6: Mannings Coefficient

261 In Contention 6, Intervenor TexPirg alleges that the drywell will fail because the water within the weir wall will not clear the first row of vents to the suppression

pool before the drywell design pressure is exceeded, because the Applicant has failed to take into account the Mannings roughness factor within the weir wall and the vents. As a result, TexPirg alleges that high pressure steam will escape into the containment without being condensed and therefore the containment vessel design pressure of 15 psig will be exceeded.*/

262 The Mannings friction factor is applicable only to gravity flows in sloping, free surface conduits. (Applicant's witness McIntyre, p. 1, following 19203; Tr. 19251; Staff's witness Fields, Tr. 19280). This factor is not applicable to the Mark III containment where the flow through the vents to the suppression pool is governed by the pressure differential between the drywell and the containment, and does not depend upon gravity flow. (Staff's witness Fields, Tr. 19280; Applicant's witness McIntyre, pp. 1-2, following Tr. 19203). However, the Applicant's vent clearing model (NEDO-20533, June 1974) takes into account the friction loses related to turning of the flow from the weir to the vent pipe,

---

*/ The Board denied the Staff's motion for summary disposition on this issue. (See September 1 Order, pp. 6-7). In opposing the Staff's motion TexPirg described this contention as its "most important safety contention. However, the Board notes that TexPirg presented no evidence nor did TexPirg's counsel appear to cross-examine Applicant and Staff witnesses on this issue. (Tr. 19205-06).

and the head losses associated with water penetration into the suppression pool. (Applicant's witness McIntyre, p. 2, following Tr. 19203; Tr. 19253). On the basis of NEDO-10320 (April 1971), the Applicant has determined that wall friction in the weir and the vents is negligible. This determination was based upon the results of calculations for the Limerick nuclear plant, reports on vent clearing times which were measured during pressure suppression tests for the Humbolt Bay nuclear plant, and comparison with the data from General Electric's Pressure Suppression Test Facility. The testimony reflects that all this data is applicable to ACNGS. (Applicant's witness McIntyre, pp. 3-5 (including Table I), following Tr. 19203; Staff's witness Fields, pp. 3-4, following Tr. 19278). The fact that the drywell consists of concrete walls is not relevant to this contention since the vent system is steel lined. (Applicant's witness McIntyre, p. 5, following Tr. 19203; Staff's witness Fields, p. 6, following Tr. 19278).

263 At page 8 of the September 1, 1981 Order the Board inquired about the margin of safety from the design of the ACNGS Mark III containment which is designed to withstand a pressure of 15 psig. The Board inquired as to the pressure at which the yield strength of the containment would be reached for the weakest component. Preliminary calculations

indicate that the ACNGS containment can withstand approximately 50 psig internal static pressure before reaching its yield strength. (Applicant's witness Lugo, p. 1, following Tr. 19204). While calculations have not been made to determine when failure would occur, it was concluded that unless the static pressure inside the containment was significantly in excess of 50 psig, there would be no containment failure. This conclusion is based upon the fact that the ultimate strength of concrete and steel usually greatly exceeds the yield strength. (Applicant's witness Lugo, p. 2, following Tr. 19204; Tr. 19259-62). With respect to potential leakage rates following a design basis accident, preoperational leakage tests will be performed to determine the actual leakage rate as compared to the maximum allowed leakage rate. (Applicant's witness Lugo, p. 2, following Tr. 19204; Staff's witness Fields, p. 23 following Tr. 19278) However, since significant material deformation is not expected when there are higher pressures in the containment, significant increases in the leakage rate should not occur for pressures less than 50 psig. (Staff's witness Fields, p. 3, following Tr. 19278).

264 The Board finds that there is no merit to TexPirg's Additional Contention 6 in that the Mannings coefficient is not applicable to vent clearing time for a design

basis accident. In any event, Applicant's vent clearing model has taken into account friction and head losses, which have been shown to be negligible.

TexPirg Contention 10: IGSCC

Doherty Contention 44: IGSCC/Water Hammer

265 TexPirg alleges in Contention 10 that Applicant has not adequately demonstrated compliance with 10 CFR Part 50, Appendix A, Criterion 31 with regard to intergranular stress corrosion cracking (hereinafter "IGSCC"). In a similar contention, Mr. Doherty alleges that the major piping components are not adequately designed to prevent propagation of intergranular stress corrosion cracks by water hammer forces.*/

266 IGSCC has been the subject of considerable study by the NRC and GE. As a result of these studies and the conclusions therefrom, the Staff issued NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," which sets forth methods acceptable to the NRC Staff to

---

*/ Applicant moved for summary disposition of these issues but the motion was denied for the reasons stated in the Order of September 1, 1981, pp. 8-13.

reduce the incidence of IGSCC in ASME Code Class 1, 2 and 3 boiling water reactor pressure boundary piping and safe ends. NUREG-0313, Revision 1, constitutes the Staff's technical resolution of the Unresolved Safety Issue A-42, "Pipe Cracks in Boiling Water Reactors." (Staff's witness Litton, pp. 3-4, following Tr. 19973).

267 The Applicant has demonstrated compliance with NUREG-0313 by specifying the use at ACNGS of materials which are not susceptible to IGSCC. Indeed, the record clearly shows that the most direct and certain solution to eliminate the potential for IGSCC in this instance is the use of materials resistant to stress corrosion.*/ These materials include plain carbon steel and low carbon grades of austenitic stainless steels. This resistence can be produced because a significant factor affecting susceptibility to IGSCC is the degree of sensitization of the steel. Sensitization is the precipitation of chromium carbide and resulting depletion of chromium level at the grain boundaries of stainless steels. Sensitization tends to decrease resistance to IGSCC and can be caused by exposure to temperatures above 800°F during welding.

---

*/ Other remedies may be used to prevent IGSCC in different components of the plant. See, for example, the findings on Doherty Contention No. 50, *supra*.

Low carbon austenitic stainless steels do not sensitize when welded and are therefore resistant to IGSCC. Plain carbon steel is a non-stainless steel alloy that has not shown susceptibility to IGSCC in BWR environments. The Applicant will use plain carbon steel or low carbon austenitic stainless steel throughout the plant where there is a potential for IGSCC. (Staff's witness Litton, pp. 4-5, following Tr. 19973; Applicant's witness Gordon, pp. 1-3 and Attachment GMG-1, pp. 1-5, following Tr. 19884; Applicant's witnesses Gunther and Malec, pp. 1-4 and Attachment LAG/WFM-1, p. 2, following Tr. 19848; Tr. 19912-13).

268 Despite great confidence in the prevention of IGSCC through the use of proper materials, ACNGS will also employ a leak detection system, which is described in PSAR Section 5.2.7 (App. Exh. 27). Since all IGSCC failures produce easily detectable leakage well before the presence of rapidly propagating cracks, this detection system provides the final conservative assurance that the safety of ACNGS will not be threatened. (Applicant's witnesses Malec and Gunther, pp. 1-2 and Attachment LAG/WFM-1, p. 2, following Tr. 19848; Applicant's witness Gordon, pp. 3-4; following Tr. 19884; Staff's witness Litton, pp. 4-5, following Tr. 19973).

269 In conclusion, the Board finds that the NRC Staff has technically resolved the generic problem of IGSCC and has developed guidelines to be followed in selecting materials which are resistent to IGSCC. The Applicant fully complies with these guidelines. Accordingly, the Board finds that there is no merit to TexPirg's contention that the Applicant has not demonstrated compliance with 10 CFR 50, Appendix A, Criterion 31 because it has failed to account for the problem of IGSCC.

270 In his Contention 44, Mr. Doherty listed a number of systems that are susceptible to propagation of IGSCC cracks due to water hammer. However, each of these systems is made of either plain carbon steel or low carbon austenitic stainless steel. Since the use of these materials effectively eliminates the potential for IGSCC in the piping, there is no basis for concern that IGSCC induced cracks will be propagated by water hammer. (Applicant's witnesses Gunther and Malec, Attachment LAG/WFM-2, pp. 1-3 and Attachment LAG/WFM-3, following Tr. 19843; Staff's witness Litton, p. 4, following Tr. 19994).

271 Aside from the specific concern raised by Mr. Doherty, the record shows that water hammer is a potential problem that can affect any fluid system. Water hammer occurs

when forces are imparted to piping from the acceleration of contained fluids. Water hammer has been designated as an Unresolved Safety Issue. As a result of that designation the Staff undertook a study to identify the cause of water hammer and to develop recommended actions to reduce its liklihood. See, NUREG-0582, "Water Hammer in Nuclear Power Plants" (July 1979). The Applicant will follow the guidance of NUREG-0582 as well as other standard industry practices in designing ACNGS to minimize the incidence of water hammer. For example, steps will be taken to increase valve closure time, piping will be arranged to preclude formation of water slugs in the steam lines and vapor pockets in water lines, control valves will be sized to be stable throughout their entire range of control, check valves will be designed to accommodate loadings resulting from valve closures following an upstream rupture, and the system will incorporate sloped lines, drains and vents to prevent water entrainment in steam lines. Where water hammer may occur, resulting forces are considered in the design of the piping system. Contrary to Mr. Doherty's contention, it is clear from the record that the Applicant is aware of the potential problem of water hammer and is taking appropriate steps to ensure that the problem will not affect plant safety.

(Applicant's witnesses Gunther and Malec, Attachment LAG/WFM-2, pp. 3-7, following Tr. 19848; Staff's witness Litton, pp. 4-5, following Tr. 19994).

272 The NRC is expected to reach its final technical resolution of this issue by December, 1982. If the resolution results in additional design changes or operating conditions not already incorporated into the design of ACNGS, the Staff will require implementation of these additional requirements. Pending such resolution, the Staff has concluded that since the probability of failure due to water hammer is low and the consequences of postulated water hammer induced accidents would be adequately limited by currently installed redundant engineered safety features, continued operation and licensing of plants can proceed with reasonable assurance that the health and safety of the public is not in danger. (Staff's witness Litton, pp. 5-7, following Tr. 19994). The Board finds that as to this issue the Staff has satisfied the requirements of Gulf States Utilities Company (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977).

TexPirg Contention 11: Flow Induced Vibration

Doherty Contention 31: Flow Induced Vibration/LPRM

273 TexPirg alleges that there is no adequate assurance that the ACNGS reactor internal structures will be able to sustain vibrations induced by operating flow transients. Intervenor Doherty's contention also raises an issue regarding the effects of flow induced vibration, but is limited specifically to the effects on Low Power Range Monitors ("LPRM's").*/

274 Applicant's witness on these issues was Mr. Martin R. Torres, who is the Manager of the Flow Induced Vibrations Group at General Electric. Staff's witness was Dr. Shou-Nien Hou, who is a Principal Mechanical Engineer in the NRC's Mechanical Engineering Branch. Both witnesses demonstrated that they had considerable knowledge concerning the problem of flow induced vibration and their testimony clearly demonstrates that the problem has been adequately resolved.

---

*/ Applicant filed a motion for summary disposition on these two contentions. The motion was denied in the Board's Order of September 1, 1980. TexPirg opposed the motion but did not appear at the hearings on these issues.

275 Mr. Torres testified that the phenomenon of flow-induced vibration has been studied extensively on previous General Electric plants. Information gathered from these studies and test programs designed specifically to study flow-induced vibration has been used to improve design and to qualify ACNGS. First, GE has developed a dynamic system analysis (described in Section 3.9.1.3 of GESSAR 238 NSSS) to define the flow-induced vibration which may result from normal reactor operation. This analysis serves two functions. GE uses it during the design and in-house testing phase of reactor internal components. The dynamic system analysis is also used to establish criteria for plant pre-operational vibration testing, discussed below. Such an analysis has been prepared for Perry Station Unit 1, the prototype 238 BWR-6 plant, and it is fully applicable to ACNGS. (Applicant's witness Torres, Attachment MRT-1, p. 2, following Tr. 19111).

276 Second, GE conducts flow tests on various reactor internals to quantify flow-induced vibration levels on the internals. These tests are conducted to verify design and are independent of vibration testing required by NRC regulations. The tests were performed at various test facilities starting as early as 1974 for the BWR-6. The tests followed procedures required by 10 CFR 50, Appendix

B. In most instances, these tests were performed using full scale, actual reactor hardware at flow rates well in excess of the operational design condition. Tests included both flow tests and, as appropriate, forced oscillation tests. Flow tests were performed on jet pumps, control rod guide tubes, low pressure coolant injection lines, feedwater spargers, fuel assemblies, in-core instrument tubes, and differential pressure lines and other components. (Applicant's witness Torres, Attachment MRT-1, pp. 3-4, following 19111).

277 Third, vibration testing requirements of Regulatory Guide 1.20 will be satisfied on a prototype plant, presently designated as Perry Unit 1, whose operation is expected to precede that of ACNGS. On the prototype plant, extensive vibration measurements will be made on major internal components during pre-operational and start-up flow testing, and an extended pre-operational flow test and inspection will detect evidence of possible undesirable effects due to vibration. The prototype test results will then be compared to those obtained from the theoretical dynamic systems analysis discussed above. The prototype tests will continue until power operating conditions are reached; however, they are scheduled to be completed prior to operation of ACNGS. In the unlikely event that the ACNGS station becomes the prototype plant, the Applicant

will follow Regulatory Guide 1.20 and perform these extensive pre-operational and operational tests. (Applicant's witness Torres, Attachment MRT-1, pp. 4-5, following Tr. 19111; Staff's witness Hou, pp. 2-3, following Tr. 19164).

278 Since ACNGS is not expected to be the prototype plant for the vibration testing requirements of Regulatory Guide 1.20, confirmatory pre-operational flow-induced vibration testing of reactor internals at ACNGS will be performed in accordance with the "non-prototype" testing provisions of Regulatory Guide 1.20. The confirmation will be made by the use of extended high-flow testing, preceded and followed by a full inspection of internals in accordance with Regulatory Guide 1.20. (Applicant's witness Torres, Attachment MRT-1, p. 5, following Tr. 19111; Staff's witness Hou, p. 2, following Tr. 19164).

279 While the extensive four-step testing analysis and inspection program is fully expected to eliminate flow-induced vibration at ACNGS, there are a number of other measures which provide added assurance and protection against flow-induced vibration at ACNGS. For example, reactor instrumentation can detect some problems long before they pose any hazard. Moreover, the ACNGS reactor will have a loose parts monitoring system. Even in the event of a failure of nonsafety components such as feedwater spargers,

the plant will still be capable of safe shutdown. (Applicant's witness Torres, Attachment MRT-1, pp. 5-8, following Tr. 19111).

280 Neither TexPirg nor Intervenor Doherty presented any evidence to support their contentions. While TexPirg alleged that Applicant had not adequately addressed the effects of flow induced vibration on a variety of reactor internals, the only alleged instances of prior damage related to feedwater spargers. However, GE has developed an improved feedwater sparger, and this new design will be used at ACNGS. The design consists of three concentric thermal sleeves, a double piston ring seal and an improved interference fit. The design has also been improved by forging the tee box junction, which reduces flow restriction and local stress concentration. It is expected that these design changes will eliminate detrimental vibration to the sparger. In any event, failure of a feedwater sparger does not affect the ability of the plant to shutdown. (Applicant's witness Torres, Attachment MRT-1, p. 7, following Tr. 19111; Tr. 19144-50, 19155-58; Staff's witness Hou, pp. 3-5, following Tr. 19164).

281 Intervenor Doherty contends that flow induced vibration has caused damage to LPRM's; however, both witnesses testified that there had never been any damage to LPRM's

which have experienced such vibration. When the LPRM vibrated in an earlier design (not the same as ACNGS), the LPRM's were not damaged. Rather the LPRM caused wear on the adjacent Zircaloy fuel channels. The cause of the LPRM vibration was due to the flow of coolant through one-inch holes in the core plate directly below the LPRM. In earlier BWR designs, these holes were not present and the LPRM did not vibrate and fuel channels were not damaged. In plants with these holes, which had subsequent fuel channel wear, the holes were plugged. The plants which had the holes plugged have shown no degradation of LPRM function and no channel wear since the design change was made in 1974. In the ACNGS design, plugging the holes was not necessary since they were not in the design. There has never been any fuel channel wear or degradation of incore instrument tubes in any BWR designed without the one-inch holes in the core plate. (Applicant's witness Torres, pp. 6-7 of Attachment MRT-1, following Tr. 19111; Tr. 19121-22; Staff's witness Hou, pp. 4-5, following Tr. 19164.

282 The Board finds that TexPirg Contention 11 and Doherty Contention No. 31 are without merit. The problem of flow induced vibration has been reviewed and tested extensively. GE has undertaken adequate testing to define the problem

and has made appropriate design changes to eliminate any problems with feedwater spargers, LPRM's and other core internals. Further assurance will be provided by the non-prototype or prototype and preoperational testing and inspection for ACNGS which has been committed to by the Applicant. The results of this preoperational vibration assurance program and test results will be submitted for NRC approval prior to the issuance of the operating license. Accordingly, the Board concludes that there is reasonable assurance that ACNGS reactor internals will be able to prevent damaging vibrations induced by operating flow transients.

TexPirg Contention 12: Cable Fires

283 In Contention 12, TexPirg alleges that the electrical wiring at ACNGS is "susceptible to fast flaming, and potential resulting common mode failures, in the event of an intense flash fire." As support for this contention, TexPirg cites a 1978 fire protection research test conducted by Underwriters Laboratory, which test TexPirg alleges indicates modification to the NRC's fire protection criteria may be necessary.

284 The Applicant established through its witnesses Montalbano and Barbieri that the ACNGS design for electrical cables meets current NRC fire protection requirements which will insure that the cables will not be subject to "fast flaming" as alleged by TexPirg. Applicant has taken certain steps to minimize both the probability and the consequences of postulated cable fires. These steps include measures to (1) prevent fires from starting, (2) detect fires quickly and extinguish those which have started, (3) insure that, should a fire occur, essential plant safety functions will not be affected. (Applicant's witness Montalbano, pp. 2-3, following Tr. 12551). First, with respect to preventing a fire from starting, the cables to be used at ACNGS will be made of materials which are fire retardant. These materials include EPR insulation and XLPE insulation with hypalon jackets. These flame retardant materials have survived the IEEE 383-1974 flame test for cable insulation. The use of these materials for the ACNGS cables demonstrate that these cables are not subject to fast flaming. (Applicant's witness Montalbano, pp. 4-6, following Tr. 12551).

285 Second, with respect to detecting and extinguishing fires quickly, the ACNGS design will use a fixed automatic PREACTION sprinkler system, an early warning fire

detection system consisting of smoke detectors located in cable tray areas. In addition, fire breaks are located in both the vertical and horizontal cable trays to prevent a postulated fire from spreading. (Applicant's witness Barbieri, pp. 7-9, following Tr. 12551).

286 Finally, with respect to precluding the spread of any postulated fire to cables which control redundant, essential plant safety functions, the ACNGS design will adhere to the tray separation distance criteria of IEEE 384-1974. As an example, the ACNGS cable vault will be designed and constructed to include a three-hour fire barrier wall, which will consist of reinforced concrete, to separate Division 1 and 4 cabling from Division 2 and 3 cabling. (Applicant's witness Montalbano, pp. 11-12, following Tr. 12551).

287 TexPirg's citation to the 1978 Underwriters Laboratory test has no relevance to ACNGS. This test was conducted to determine the effects of fire on a vertical cable tray arrangement with cables protected by a mineral wool fire barrier. The use of mineral wool blankets as a fire barrier will not be used at ACNGS. (Applicant's witness Montalbano, p. 7, following Tr. 12551).

288 While the Applicant has demonstrated that its current design meets applicable NRC requirements and assures that

no fast flaming of cables will occur, the NRC has published additional fire protection requirements in 10 CFR Part 50, Appendix R. Appendix R does not on its face apply to ACNGS since it specifically states that it is only applicable to plants licensed to operate prior to 1979. However, the Staff testified that it intends to apply the requirements of Appendix R to ACNGS by revising its Branch Technical Position 9.5-1 to include these requirements. (Staff's witness Harrison, p. 2, following Tr. 12647). The Staff will review ACNGS against these requirements at the operating license stage and the ACNGS safe shutdown capability for fire protection will be judged against the acceptance criteria outlined in Section III.G of Appendix R to 10 CFR Part 50. (Staff's witness Harrison, p.2, following Tr. 12647; Tr. 12648). Applicant agrees that if Appendix R requirements are made applicable, it will review its design in light of those requirements and address those requirements at the operating license stage. (Applicant's witness Barbieri, Tr. 12553).

289 Based on the foregoing findings the Board finds that the Applicant has established that its cables will not be subject to fast flaming contrary to the allegations of TexPirg in Contention 12.

TexPirg Additional Contention 21: Occupational Exposure.

290 In Supplement No. 1 to the Final Environmental Statement (Staff Exh. 12), the NRC Staff projected an occupational dose of 500 man-rems per year for ACNGS based upon data from operating light water reactor plants similar in size and type. (See Staff Exh. 12, p. 5-29). In Additional Contention 21, TexPirg alleges that projected radiation exposure to workers at ACNGS has been significantly underestimated. In support of its contention TexPirg cites an article which appeared in Nuclear Engineering International (February 1979, p. 36) and which indicated that certain light water reactors experienced actual occupational exposures higher than the average annual dose originally projected by the Staff.

291 While the basis for TexPirg's contention may be factually correct (Staff's witness Nehemias, p. 2, following Tr. 12376), the conclusion TexPirg would reach, i.e., that the estimated occupational exposure at Allens Creek is underestimated, does not follow. First, the recorded doses set forth in the article cited by TexPirg, which are based upon the ratio between radiation dose and electricity generated (man-rem/GW(e) year), are not comparable to the estimates in Supplement No. 1, which

are based on actual recorded annual doses expressed in man-rem per reactor year. As Applicant's witness Baron pointed out, the differences in the units of measurement renders pointless any comparison between the two. Specifically there is an inverse relationship between power production and accumulated dose, with the higher worker doses coming when a plant is out of operation for maintenance or modification. (Applicant's witness Baron, pp. 2-3, following Tr. 12266; Staff's witness Nehemias, Tr. 12481).

292 Second, the data used by Staff in making its estimate for ACNGS was based upon actual figures from operating plants. Although the recent average annual reactor dose at BWR's has been about 740 man-rems per reactor year, particular plants have experienced annual average doses as high as 1850 man-rems and as low as 183 man-rems. This wide range of doses is the result of a number of factors including the amount of routine and special maintenance required at a particular plant. (Staff's witness Nehemias, p. 3, following Tr. 12376).

293 Due to a concern over increased worker doses during plant maintenance, modification and daily operations, the NRC in 1975 adopted new requirements and guidance designed to reduce occupational exposure at nuclear plants to levels "as low as reasonably achievable" (ALARA). Plants designed,

constructed and operated in accordance with the ALARA philosophy should experience lower exposure rates than older plants where such considerations were not given similar emphasis. (Staff's witness Nehemias, p. 6, following Tr. 12376; Applicant's witness Baron, p. 3, following Tr. 12266).

294 In its review of ACNGS, Staff found that the Applicant is committed to design features and operating practices that will assure that individual and collective occupational radiation doses are maintained within the limits of 10 C.F.R. Part 20 and are as low as is reasonably achievable. (Staff's witness Nehemias, p. 5, following Tr. 12376). Indeed, HL&P has an ALARA training program and instructs the engineering and health physics members of the ACNGS team to follow a manual containing ALARA guidelines in evaluating plant design features. (Applicant's witness Baron, p. 6, following Tr. 12266). Both Ebasco, the Architect-Engineer, and General Electric, the NSSS supplier, are involved in the ALARA design process. This process has resulted in a number of changes in the design of ACNGS and these changes will most likely result in lower worker exposure. (Applicant's witness Baron pp. 8-15, following Tr. 12266).

295 As a result of the incorporation of ALARA philosophy into the design and operation of newer BWR's, it is Staff's opinion that occupational exposures will be less for such plants as a group than those experienced by the older plants now operating and upon which Staff's Supplement No. 1 estimate was based. (Staff's witness Nehemias, p. 6, following Tr. 12376; Tr. 12402.) In this regard, Applicant's witness Baron explained that as a result of the ALARA design changes, exposure levels at Allens Creek will be less than the 500 man-rems per reactor year estimate set forth in Supplement No. 1. (Applicant's witness Baron, p. 3, following Tr. 12266). Dr. Nehemias testified that any revision of the estimate of average annual occupational radiation exposure based on recent operating experience would not change any of the Staff's conclusions in the FSFES. (Staff's witness Nehemias, pp. 6-7, following Tr. 12376).

296 The Board finds that there is no evidence in the record to support TexPirg's contention that the estimate of worker exposure appearing in supplement No. 1 to the FES is significantly understated. Based on Applicant's commitment to design the plant in accordance with ALARA principles, it is reasonable to expect that worker exposure at Allens Creek may in fact be below the Staff estimate of 500 man-rems per reactor year.

TexPirg Additional Contention 26: Computer Code Error

297 TexPirg's Contention 26 alleges that the computer program used to calculate the stresses on the reactor and containment during the design basis earthquake is defective because it subtracts forces when they should be added. Through discovery, TexPirg disclosed that this contention was based solely on problems identified in a Stone & Webster computer code (named "SHOCK II") that had been applied to five plants designed by Stone & Webster. (Applicant's witness Korde, pp. 1-2, following Tr. 11255). The Applicant does not use the SHOCK II pipe stress computer program. Rather, it will use the PIPESTRESS 2010 computer program developed by Ebasco. (Applicant's witness Korde, pp. 2-3, following 11255). Contrary to TexPirg's allegation, the PIPESTRESS program does not add or subtract responses algebraically. The responses are combined by the square root of the sum of the squares. (Applicant's witness Korde, p. 3, following Tr. 11255; Tr. 11269). This has eliminated the concern raised with respect to the SHOCK II program. (Applicant's witness Korde, p. 3, following Tr. 11255). This method of analysis complies with Regulatory Guide 1.92, "Combining Nodal Responses and Spatial Components in Seismic Response Analysis." (Staff's witness Hartzman, p. 2, following

Tr. 11285). Compliance with this Regulatory Guide assures that the Applicant's computer program combines response components in an acceptable manner and has been so verified through comparison with other programs and hand calculations. (Staff's witness Hartzman, p. 4, following Tr. 11285). The Applicant's pipe stress analysis program has been benchmarked against other programs, will be checked against sample problems contained in NUREG/CR-1677, "Piping Benchmark Problems, "Vol. 1 (Aug. 1980), and selected analyses will be independently verified by the Staff prior to issuance of the operating license. (Applicant's witness Korde, p 3, following Tr. 11255; Staff's witness Hartzman pp. 3-4, following Tr. 11285).

298 The Board finds that the Ebasco computer program, described above, does not have the deficiencies alleged in the contention, regardless of whether such deficiencies existed in the SHOCK program.

TexPirg Additional Contention 28: Post Accident Monitoring

299 TexPirg contends that the ACNGS control room design and the post-accident display instrumentation are not sufficient to ensure that the operators can safely control the plant under accident conditions. TexPirg alleges that the TMI

incident demonstrated that operators may make mistakes because of defective instruments or their location in the control room.

300 TexPirg never explained nor proved what instruments were defective at TMI nor how any such defects could relate to ACNGS. Furthermore, TexPirg presented no evidence of any comparability between the control rooms for TMI and Allens Creek. In fact, the control room at TMI bears no resemblance to the Allens Creek control room. (Tr. 15981-2). Allens Creek will use General Electric's NUCLENET/1000 Control Complex. This control complex incorporates the use of computers and cathode ray tubes. (Applicant's witness Ranzau, p. 14, following Tr. 15783; Tr. 1955-60). This is the most advanced state-of-the-art design, and the NUCLENET/1000 is considered to be one of the best control complexes in any nuclear plant that has been reviewed by the NRC Staff from the standpoint of human factors engineering and design. (Staff's witness Schemel, pp. 3-5, following Tr. 16438; Tr. 16548-50; 16569-79).

301 In addition to the use of advanced control systems, the NUCLENET complex has been carefully designed to assure that the location of instruments satisfy human factors principles. The design process included an analysis of

all functions necessary to operate the plant safely, an allocation of functions between operator and machine, and a qualitative verification of the functional allocation. In designing the NUCLENET complex, General Electric assembled a design team which included experts in controls and control systems design, computer technology, industrial design, operator training, power plant testing and operations, and behavioral science. The premise upon which the design is based is that optimum control is achieved when there is an allocation of control functions between the operator and machine which recognizes that each performs certain functions better than the other, and that, once the allocation is made, the design permits efficient and effective manipulation of controls by the operator. (Applicant's witness Ranzau, pp. 14-15; following Tr. 15783; Tr. 15951-54, 15978-81; Staff's witness Schemel, pp. 3-4; following 16438)

302 As a result of the TMI incident, the NRC has developed new regulatory guides that apply to control room design. See, NUREG/CR-1580 and NUREG-0659. Following the TMI incident, HL&P formed a Control Room Evaluation Task Team to perform a preliminary assessment of the ACNGS Unit 1

control room to see if further improvements were necessary in light of the NRC's new guidelines. HL&P built a full size mockup of the front row panels of the NUCLENET/1000 Control Complex. A human factors engineering evaluation was performed on the mockup. The control room was evaluated as is, without regard to planned changes in layout and changes required as a result of Three Mile Island. The evaluation consisted of the application of a human factors engineering design check-list developed and prepared by the BWR Control Room Owners Group following the TMI incident. Several design changes were made and further enhancement activities are now underway. (Applicant's witness Ranzau, pp. 15-16, following Tr. 15783; Tr. 15986-95; Tr. 16005-06; Tr. 16054-56). In addition, HL&P added a Safety Parameter Display System (SPDS). The SPDS will display the full range of important plant parameters and will indicate when plant parameters are approaching or exceeding process limits. (Applicant's witness Ranzau, pp. 16-17, following Tr. 15783; Tr. 16007-17; Staff's witness Schemel, p. 2, following Tr. 16438; Tr. 16587-89).

303 Although the final design of the control room will be submitted to the NRC for review at the operating license stage (and prior to construction of the control panels), the Staff has determined that the currently designed

NUCLENET/1000 Control Complex complies with the NRC requirements pertaining to control room design. (Staff's witness Schemel, p. 4, following Tr. 16438; Tr. 16481-2).

304 Safety related instrumentation included in the ACNGS control room (including the SPDS) will be designed to meet the reliability requirements outlined in Reg. Guide 1.97, Rev. 2 and Section II.F.3 of NUREG-0718, as documented in the ACNGS PSAR (App. Exh. 27), Appendices C and O. Contrary to TexPirg's contention, these commitments provide reasonable assurance that ACNGS will be equipped with reliable, state-of-the-art instrumentation. (Applicant's witness Ranzau, p. 17, following Tr. 15783).

305 The Board finds that the ACNGS control room is a state-of-the-art design and further finds that that Applicant has taken all necessary steps to assure proper design of the control room in accordance with NRC requirements. The Board finds that the control room at ACNGS bears no relationship to the control room at TMI, and thus, TexPirg's contention is without merit. The control instrumentation applies state-of-the-art human factors principles in design and layout. We conclude that there is reasonable assurance that the ACNGS control room design and post-accident display instrumentation will be adequate to ensure that operators can safely control ACNGS under accident conditions.

TexPirg Additional Contention 31: Technical Qualifications

306 In this contention, TexPirg alleges that HL&P is not technically qualified to construct ACNGS because of several specific construction problems at the South Texas Project (STP) nuclear plant.

307 TexPirg presented no evidence on this contention and did not appear to cross-examine Applicant's and Staff's witnesses. Applicant presented two witnesses: Mr. George W. Oprea, Executive Vice President for Nuclear Projects, and Mr. Jerome H. Goldberg, Vice President for Nuclear Engineering and Construction. These are the two most senior executives responsible for the design and construction of ACNGS. Staff also presented two witnesses: Mr. Frederick R. Allenspach, who was responsible for reviewing HL&P's organizational structure and technical qualifications, and Mr. John W. Gilray, who was responsible for reviewing HL&P's quality assurance and quality control (QA/QC) program. The testimony of these witnesses establishes that HL&P is technically qualified to construct Allens Creek.

308 The South Texas Project is a two-unit nuclear plant currently under construction. STP is jointly owned by four electric

utility systems, and HL&P is the Project Manager. (See App. Exh. 19). The original architect/ engineer for STP was Brown & Root, Inc. On April 30, 1980, the NRC fined HL&P $100,000 and issued an order requiring HL&P to show cause as to why construction should not be halted because of construction problems at STP. (Doherty Exh. 4; Tr. 18286-87). This order raised a number of construction problems at STP beyond those cited in TexPirg's contention. Neither Applicant nor Staff sought to dispute the fact that there have been construction problems at STP. (Tr. 18328). However, they emphasize, and correctly so, that the key inquiry here is whether HL&P has improved its organizational structure and management capability in order to prevent a recurrence of similar problems at ACNGS. (Tr. 18351-58; 18481-84). Applicant did address the specific allegations in TexPirg's contention and demonstrated that none of those allegations taken individually, or as a whole, reflect serious doubt on HL&P's technical competence to design and construct ACNGS. (Applicant's witnesses Oprea and Goldberg, pp. 14-24, following Tr. 18084; Tr. 18247-304). Most important, however, is the fact that the instances cited in the contention all took place prior to the NRC's show cause order on STP. There have been substantial changes in HL&P's organizational structure and management capability following the show cause order.

309 Subsequent to the problems identified at STP, HL&P implemented a major reorganization to provide better management control of its nuclear projects. Specifically, HL&P changed its organization so that a senior corporate officer, Mr. Oprea, is now in charge of the nuclear program. He is responsible for all nuclear activities within HL&P related to design, engineering, construction, nuclear fuel, operation and quality assurance and he reports directly to the President and Chief Executive Officer of HL&P. Prior to this organization change, Mr. Oprea was responsible for both nuclear and fossil projects. He is now free to devote his attention almost exclusively to nuclear matters pertaining to the design and construction of both STP and ACNGS. Within the new organization, HL&P has established the new position of Vice President of Nuclear Engineering and Construction. This position has been filled by Mr. Goldberg, who has substantial experience in designing and constructing nuclear power plants. Furthermore, HL&P has created a new position of Vice President of Nuclear Operations. This position has been filled by the hiring of Mr. Jerrold G. Dewease who has had substantial experience in the area of nuclear power plant operations and maintenance. (Applicant's

witness Oprea, p. 5, following Tr. 18084; Tr. 18092-3, 18133-42, 18378-81, 18392-96; Staff's witness Allenspach, p. 3, following Tr. 18417; Staff's witness Gilray, Tr. 18422-34).

310 The organization reporting to the Vice President of Nuclear Engineering and Construction consists of the Manager South Texas Project, Manager ACNGS, Manager Nuclear Services, and Manager Nuclear Licensing. The Manager ACNGS has responsibility for the design, procurement, and construction at ACNGS. Thus, control of the architect/engineer (A/E) and the designer and fabricator of the nuclear steam supply system (NSSS) at ACNGS is totally separated at the project level from these activities at STP. This is a typical and desirable method of organization. (Applicant's witness Goldberg, pp. 8-10, following Tr. 18084; Staff's witness Allenspach, p. 4, following Tr. 18417; Tr. 18459-60). Under the Manager ACNGS, HL&P has an in-house staff of engineers and managers to oversee the design and verify conformance with the applicable regulations, codes and other design criteria for ACNGS. At this time, sufficient manpower is maintained to meet current responsibilities of the project. Additionally, in some specific cases, temporary engineering support is assigned from line departments within the Company, or consultants are contracted to work under the

direction of HL&P personnel. In this regard HL&P has an extensive corporate manpower pool to draw upon when necessary. It has approximately 250 employees in various disciplines including management, nuclear, civil, electrical, and mechanical engineering, health physics and nuclear fuels. Project staffing on ACNGS will increase commensurate with the project activities that are planned or underway. (Applicant's witness Oprea, pp. 6-7, following Tr. 18084; Tr. 18151-55).

311 As frequently occurs within the industry, HL&P will place primary responsibility for the design and construction of ACNGS on the A/E, which in this instance is Ebasco. (Applicant's witnesses Oprea and Goldberg, Tr. 18122-23). The responsibility for doing the job correctly rests initially with the A/E; therefore, the technical competence of Ebasco is of considerable importance. (Applicant's witnesses Oprea and Goldberg, Tr. 18228-30; 18384-18392). As is noted numerous times in the record here, many of the problems at STP are attributable to Brown & Root's relative inexperience in designing and constructing nuclear plants. (Applicant's witnesses Oprea and Goldberg, Tr. 18098-99, 18102-10). In contrast, Ebasco has provided engineering, construction and consulting services to utilities in the United States and throughout the world for 70 years. In the past 18 years, Ebasco has been the

A/E on 25 nuclear projects; fourteen of which are General Electric BWR's. Currently, Ebasco maintains a permanent force of approximately 4,000 personnel to carry out services related to power generation, transmission and distribution. Approximately 1000 of these employees are specifically identified with nuclear activities. Thus, Ebasco Services has substantial experience in the design and construction of BWR nuclear plants. There is simply no reason to expect that the problems encountered at STP by Brown & Root will occur at ACNGS. If they do, HL&P is capable of recognizing those problems and taking corrective actions. (Applicant's witness Oprea, pp. 20-21, 22-24, following Tr. 18084).

312 It is also important to consider the technical competence of General Electric. The General Electric Company is one of the major designers and fabricators in this country of nuclear reactors and nuclear fuel. It has been in the nuclear business since 1955 and presently has over 80 reactors either complete, under construction, or on order. Twenty-four of these reactors are BWR/6 design with Mark III Containments. GE has extensive research and development facilities which are used in the design of the BWR. These facilities have both full scale and scaled down models and cover such topics as core thermal hydraulics, mechanical testing of BWR material, performance

characteristics of various BWR components, blowdown loads on containment, and servicing and maintenance of the BWR. Thus, GE has extensive experience, knowledge and capability to design and fabricate the BWR. (Applicant's witness Goldberg, pp. 24-25, following Tr. 18084).

313 Notwithstanding the technical competence of GE and Ebasco, HL&P is also committed to developing a highly skilled and qualified staff. (Applicant's witnesses, Oprea and Goldberg, Tr. 18150-58, 18370-75). It is clear that HL&P has learned a great deal from its experience at STP. It now has a strong management commitment to the QA/QC program (Staff's witnesses Allenspach and Gilray, Tr. 18486-92), and it has made numerous beneficial changes in its QA/QC program. One of the most significant changes related to the removal of several layers of management in the quality assurance/quality control (QA/QC) organization in order to put Mr. Oprea in direct contact with the QA/QC personnel. This has resulted in a tighter communication link between executive management and the QA/QC field personnel. (Applicant's witness Oprea, Tr. 18088-90; 18396). Additionally, the overall programs of HL&P have been revised to include improved controls and procedures to provide a more effective management system in carrying

out the program. These controls include (a) increasing the QA and QC organizations' responsibilities and authority in day-to-day site activities and in stop work actions; (b) increasing the training and qualification programs for site personnel; (c) modifying the system for identifying nonconformances and initiating and verifying corrective actions to assure timely and effective control of deficiencies; (d) establishing a trend analysis program for systematic review of nonconformances, corrective actions, audit deficiency reports, engineering design deficiencies, and vendor nonconformances to determine where investigative and corrective actions are necessary; (e) improving the field design change program by increasing and assigning qualified engineering staff at the site for the timely analysis and independent review and verification of design changes; (f) increasing as-built control of the design change activity at the site; and (g) revising the audit programs by increasing the auditing skills of the audit staff and by enhancing the audit activity in the review of records and direct observations of work being performed to assure procedural adherence and compliance with the QA program. (Staff's witness Gilray, pp. 4-5, following Tr. 18417; Applicant's witnesses Oprea and Goldberg, Tr. 18088-99, 18395, 18235-38).

314 Coincidentally, Applicant has demonstrated that its QA/QC program complies with new TMI related requirements. These new requirements are set forth in Item I.F.2 of NUREG-0718. Applicant's reply to NUREG-0718 is contained in PSAR, Appendix O (App. Exh.27). The Staff has concluded that the HL&P QA program and the Ebasco and GE QA programs meet the requirements of NUREG-0718. All of their QA programs satisfy the considerations of: (a) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (b) performing quality assurance/ quality control functions at construction sites to the maximum feasible extent; (c) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (d) establishing criteria for determining QA programmatic requirements; (e) establishing qualification requirements for QA and QC personnel; (f) sizing the QA staff commensurate with its duties and responsibilities; (g) establishing procedures for maintenance of "as-built" documentation; and (h) providing a QA role in design and analysis activities. (Staff's witness Gilray, p. 5, following Tr. 18417; Tr. 18445-47; SER Supp. 3, pp. II-25 to II-30 (Staff Exh. 20).

315 The Board finds that the Applicant has qualified and experienced executives in key management positions. The organization structure below these executives, as described in Chapter 17 of the PSAR, is properly organized and staffed with qualified and experienced personnel to supervise the design and construction of Allens Creek. Clearly, both Ebasco and GE have the technical capability and experience to design and construct Allens Creek. Applicant's QA/QC program, as described in Chapter 17 of the PSAR complies with 10 CFR Part 50, Appendix B, and NUREG-0718. The record demonstrates that Applicant has gained considerable experience from its involvement at STP and that this has led to an overall strengthening of its management structure and capability. Accordingly, the Board finds that Applicant is technically qualified to construct ACNGS.

TexPirg Additional Contention 34: Hydrogen Monitoring

316 TexPirg contends that the design of ACNGS should include a monitor to detect the occurrence of a hydrogen explosion. In support of the contention TexPirg argues that

a hydrogen explosion occurred at Three Mile Island ("TMI").*/

317 TexPirg did not appear at the hearing on this issue and presented no evidence to prove that there was in fact a need to monitor for hydrogen explosions. Furthermore, there is no requirement in NUREG-0718 which would require installation of such monitors. (Staff's witness Fields, Tr. 19388-92). As discussed in the findings on Board Question 15, infra, this NUREG establishes the TMI related requirements for plants at the CP stage. In any event, there has been no demonstration by TexPirg that any such monitors would be needed notwithstanding the requirements of NUREG-0718.

318 As described in the findings on Board Question 4A, infra, ACNGS is being designed to prevent hydrogen concentrations in the containment from ever reaching detonable limits following a LOCA. Under the regulations that existed prior to TMI, the Applicant was required to control hydrogen after a LOCA in accordance with 10 CFR § 50.44. Applicant has designed a hydrogen recombiner system

---

*/ The Staff moved for summary disposition of this issue but the motion was denied for the reasons stated in the Order of September 1, 1980, pp. 20-22. TexPirg did not appear at the hearings on this issue despite its opposition to the summary disposition motion.

to handle the amount of hydrogen that is assumed to be present following a LOCA. NUREG-0718 requires that this protection be carried even further. Applicant is now committed to provide a hydrogen control system that would adequately handle hydrogen levels that would be produced from a metal-water reaction of 100% of the active fuel clad. The net effect is that ACNGS will have sufficient hydrogen control mechanisms in place to prevent hydrogen from ever reaching a detonable concentration. (Staff's witness Fields, Tr. 19399-401; Applicant's witness Hucik and Weingart, Tr. 20366-67). Thus, the Board finds that there is no reason to require Applicant to install instruments that would detect a hydrogen explosion.

319 In our Order of September 1, 1981, denying a motion for summary judgment on this issue, the Board recognized that the foregoing conclusion depended in substantial part on the design of the recombiners. Accordingly, the Board posed several questions to be answered during the hearings. Those questions, and the findings related thereto are as follows:

1. Supply test results supporting the adequacy of the type and size of thermal recombiners to be used.

320 The recombiners currently planned for installation inside the ACNGS containment are Westinghouse thermal recombiners with a flow capacity of 100 SCFM. The Westinghouse recombiner design has been thoroughly tested to assure its performance during post LOCA conditions. The test results confirm that the hydrogen recombiner of the size and type to be used at ACNGS will perform as indicated on PSAR Figure 6.2-29 (App. Exh. 27). (Applicant's witness Weingart, pp. 6-7, following Tr. 20341; Staff's witness Fields, p. 2, following Tr. 19374; Tr. 19408).

2. Discuss the effects of poisoned recombiner surfaces and convective circulation in reducing recombiner effectiveness.

321 The recombiner has been exposed during testing to severe environmental effects such as steam, containment spray, radiation and temperature without any degradation of performance. In actuality, the recombiner will be enclosed and thus protected from impingement by containment spray. Moreover, there are no catalysts employed which could be degraded by any "poisoning" process. (Staff witness Fields, p. 3, following Tr. 19374; Applicant's witness Weingart, p. 6, following 20341).

322 Regarding convective circulation, no stratification or pocketing of hydrogen is expected because of various factors present inside the containment such as

heat sources, heat sinks and containment sprays. PSAR Section 6.2.5.3.3 (App. Exh. 27) adequately describes the various analyses performed to demonstrate drywell and containment hydrogen mixing, and the hydrogen redistribution from the drywell to the containment due to the operation of the Drywell-Containment Hydrogen Mixing Subsystem. (Staff's witness Fields, p. 3, following Tr. 19374; Applicant's witness Weingart, p. 5, following 20341; Tr. 20372; Applicant's witness Hucik, pp. 3-5; following Tr. 20297; Tr. 20342-45; Staff's witness Fields, Tr. 19445-52).

3. Provide sufficient recombiner dynamic analysis to demonstrate that 3% concentration of hydrogen is a conservative alarm set-point.

323 Applicant presented an analysis showing the typical hydrogen concentration time history in a Mark III Containment following a recirculation line design basis accident. The analysis is based on the very conservative assumptions of Regulatory Guide 1.7. At the time when the containment hydrogen concentration reaches 3% ( 17 days), the rate of hydrogen evolution from the suppression pool due to radiolysis is less than 1 SCFM. That translates to a hydrogen concentration rise of 0.1%/day. With a nominal recombiner warm-up time of 3 hrs. there is more than enough time for the operator to activate a back-up system in case one fails. (Applicant's

witness Hucik, pp. 4-5, following Tr. 20297; Tr. 20346-48).

4. Describe the relationship - functional and geometrical - between the alarm sensor and the eight monitoring samplers.

324 The ACNGS Hydrogen Monitoring Subsystem, which is designed to the requirements of Regulatory Guide 1.7, will have the ability to obtain samples from various locations within the drywell and the containment. These points are selected to provide complete coverage of the drywell and containment. The system consists of two identical analyzer trains each powered from a different emergency bus, and each having the ability to monitor any of the sample points. Redundant connections will be provided at each sampling location (one for each analyzer). The redundant analyzer equipment will be located in the Reactor Auxiliary Building approximately 135° apart. Readouts and control capability will be provided in the Control Room. The hydrogen monitoring system includes sample and return lines, isolation valves, hydrogen analyzers and sample pumps. The equipment excluding the isolation valves and piping is located in the Reactor Auxiliary Building. Each sample location can be monitored by either analyzer through a sample selection manifold. The hydrogen concentration is determined in the analyzer and the volume percent is recorded in the Control Room. The analyzer

has a range of 0-5% hydrogen with an accuracy of ± 2.0% of full scale and a minimum sensitivity of 0.2% hydrogen by volume. The concentration is recorded during sampling and an alarm is automatically actuated if the concentration at any sample point exceeds 3.0% by volume. The hydrogen monitoring system is manually actuated from the control room within 30 minutes of a safety injection signal. If Regulatory Guide 1.7 assumptions are used in the generation rates of hydrogen, operator action, and thus hydrogen monitoring, is not needed for up to 9 hours after a LOCA. (Staff's witness Fields, pp. 5-6, following Tr. 19374, ; Applicant's witness Weingart, pp. 3-4, following Tr. 20341).

5. Describe the ability to periodically test the operability of the monitoring, alarm and recombiner systems.

325 The hydrogen monitoring and alarm system can be tested and calibrated by introducing low concentration $H_2$ and $N_2$ mixtures for zero adjustment and scale calibration. This calibration can be completed from the control room. The recombiners have the capability to be periodically energized to confirm their operability requirements. These tests will be performed at the power levels needed to perform their function of recombining hydrogen with oxygen and for a long enough period to demonstrate stability

of the system. (Staff's witness Fields, p. 6, following Tr. 19374; Applicant's witness Weingart, pp. 4, 7, following Tr. 20341).

6. State the basis for confidence that pockets of high hydrogen concentration will not elude the monitoring and alarm systems.

326 As described above, the mixing mechanisms assure that there will be no pocketing of hydrogen inside containment. Moreover, hydrogen monitors will be located wherever hydrogen could possibly collect. (Staff's witness Fields, p. 6, following Tr. 19374; Applicant's witness Weingart, p. 5, following Tr. 20341; Tr. 20371-72; Applicant's witness Hucik, p. 3-4, following Tr. 20297.

7. Describe the nature of the backup containment hydrogen purging system that may be required to function at a time when the containment atmosphere is radioactive.

327 The backup containment hydrogen purge system consists of a supply line and an exhaust line that are used to exhaust the hydrogen mixture from the containment to the Shield Building Annulus. After being recirculated in the annulus to allow for radioactive decay, the gas would then be released through the Standby Gas Treatment System to the environs. (Staff's witness Fields, p. 7, following Tr. 19374; Applicant's witness Weingart p. 7, following Tr. 20341).

328 Based on the foregoing answers to the Board's questions, the Board concludes that the hydrogen recombiners will adequately perform their intended function.

TexPirg Additional Contention 36 (McCorkle 17):
Charcoal Adsorber Fires.

329 TexPirg contends that the Applicant should provide water sprays as a fire protection measure for the charcoal adsorber beds in the off-gas system. The basis for this contention is TexPirg's belief that such a water spray system is required by Regulatory Guide 1.52. In addition, TexPirg cites an accident at Browns Ferry in 1977 as further support for its contention.

330 The charcoal adsorbers will be utilized in the Off-Gas Processing System, the ECCS Area Filtered Exhaust System (AFES), the Standby Gas Treatment System (SGTS), and the Control Room Emergency Filtration System (CREFS) at ACNGS. The Off-Gas Processing System processes air and noncondensible gases drawn from the main condenser by a steam air ejector. The system utilizes four (4) charcoal adsorber tanks operating at a constant temperature of 40°F cooled by a glycol gas cooler. This system is not an Engineered-Safety-Feature (ESF) Atmosphere Cleanup system and therefore, Regulatory Guide 1.52 is not applicable. The system meets, however, the NRC Effluent Treatment Systems Branch Technical Position 11-1, Rev. 1, criteria for seismic design and quality group classifi-

cation. (Staff's witness Lee, p. 2, following Tr. 12464). The ECCS-AFES, the SGTS and the CREFS are classified as ESF Atomosphere Cleanup Systems and are designed to control the releases of radioactive materials in gaseous effluents following a design basis accident (DBA). These ESF Atmosphere Cleanup Systems, including the charcoal adsorbers, will be designed in accordance with the guidelines of Regulatory Guide 1.52, Revision 1. (Id.).

331 In order to cool the charcoal adsorbers following a DBA, the Applicant will use a system design which includes provisions for preventing adsorber fire by ensuring continued cooling of charcoal adsorbers in the ESF Atmosphere Cleanup Systems by low-flow air bleed. This is accomplished by providing cross connections between redundant air bleed fans and the ESF filter trains. (Id.).

332 The charcoal adsorbers which appears to be the main focus of the contention is in the Off-Gas System. This system uses four charcoal adsorber tanks, each containing approximately seven tons of charcoal, operating at a constant temperature of 40°F and cooled by a glycol gas cooler. The charcoal adsorber vessels are designed to adsorb and delay the xenon and kryton isotopes from the

carrier gas stream. (Applicant's witness Weingart, p. 2, following Tr. 12417; Staff's witness Lee, pp. 2-3, following Tr. 12464).

<u>333</u> A refrigeration system provides cooling both for the charcoal adsorber vessels and the incoming gas stream. In addition, each of the four vessels has three heat sensors which would detect the rise in temperature caused by a charcoal fire. When a high temperature is indicated in the charcoal beds, they can be isolated and the charcoal adsorber vessels purged with nitrogen. The nitrogen would displace the air in the vessels and extinguish the fire. (Applicant's witness Barbieri, p. 3, following Tr. 12417).

<u>334</u> The system to be installed at Allens Creek is similar to the system at Browns Ferry. In 1977 a deflagration wave there apparently caused isolated regions of the charcoal adsorber beds to burn. The beds were subsequently isolated from the off-gas flow and purged with nitrogen to stifle combustion and restore temperature. The record indicates that following that incident there was no notable increase in stack gas activity. (Applicant's witness Barbieri, p. 4, following Tr. 12417; Staff's witness Lee, p. 6, following Tr. 12464). Indeed, Mr. Lee testified that the radiological consequences of a

fire in the charcoal adsorber beds in the Off-Gas System would not be significant from the standpoint of public health and safety. (Staff's witness Lee, Tr. 12477).

335 The testimony of the witnesses for both Staff and Applicant demonstrates that the charcoal adsorber system proposed for the ACNGS Off-Gas System provides adequate, dependable and proven protection against fire. The Board finds that the addition of a water spray system as suggested by TexPirg is not only unnecessary and not required, but could, in fact, be counterproductive due to problems associated with such a system's inadvertent actuation that are not present in the current system. (Applicant's witness Barbieri, p. 2 following Tr. 12417).

TexPirg Additional Contention 38: SDV Float Switch

336 The Board granted summary disposition of this contention in the Order of September 1, 1981. However, in that Order the Board asked the following questions regarding the differential pressure transmitters:

1. What environmental qualification testing has been or will be conducted on the proposed solid state differential pressure level transmitter system, with what results, and on what schedule?

2. Will reliability and operational test information be required before final acceptance of this design departure from earlier systems, of what nature, and on what schedule?

337 As to the first question, the transmitters used for measuring the scram discharge volume level at Allens Creek will be qualified in accordance with the recommendations of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." (Applicant's witness Heidt, p. 1, following Tr. 20610). Category I requirements represent the Staff's interpretation of IEE Standard 323-1974, "IFE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." (Staff's witness Kennedy, p. 2, following Tr. 20648). These documents require that the transmitters be demonstrated to be operable during and after exposure to the environment caused by a design basis accident at the end of their qualified life. (Id.). In fact, numerous qualification tests in accordance with NUREG-0588 Category I requirements are presently being conducted by the industry to achieve full qualification of all safety-related equipment by June 30, 1982 (the Commission imposed deadline) or, in the case of Allens Creek, at the operating license stage of review. By the time that stage is reached, therefore, there will have been

transmitters reviewed and approved by the Staff which are fully qualified to the above requirements. (Staff's witness Kennedy, p. 4, following Tr. 20648).

338 As to the second question, the use of the solid state transmitters is part of the package of design improvements to the scram discharge volume that were developed during the design evolution of the more modern, solid state safety system for the GE BWR-6 plants; (Applicant's witness Heidt, pp. 1-2, following Tr. 20610). The use of the differential pressure measurement method proposed by the Applicant has been used extensively in both BWR's and PWR's to measure level and to provide input for safety functions for reactor vessel and pressurizer level as well as various flow measurements. (Staff witness Knight, p. 4, following Tr. 20648). Therefore, no special reliability and operational test information beyond that described above has been requested by the Staff. (*Id.*).

Reliability and operability of this system will be assured by monthly testing in accordance with Standard Technical Specifications. Technical Specification reporting requirements will identify any frequently occuring failures in the level sensing system whether found during functional testing or observed during daily channel checks. (Staff's witness Knight, p. 5, following Tr. 20648). Additionally, operational experience will be gained during the preoperational test phase for ACNGS and from other plants which will operate prior to ACNGS. (Staff's witness Knight, p. 5, following Tr. 20648; Applicant's witness Heidt, pp. 1-2, following Tr. 20610).

339 Based upon the testimony summarized above, the Board finds that sufficient measures are being taken to assure reliable scram discharge volume monitoring for the Allens Creek facility.

TexPirg Additional Contention 39:
Generic Issue A-11 Fracture Toughness

340 In Contention 39, TexPirg alleges that ACNGS should not be licensed until the generic safety issue described in NUREG-0371 and designated as Task A-11, which involves reactor vessel materials toughness, has been resolved.

341 Task A-11 relates to the potential for reactor pressure vessel failure as a result of a reduction in vessel material fracture toughness due to long term neutron irradiation. Both the Applicant and Staff witnesses testified that the problem stated in Task A-11 only related to older operating pressurized water reactors.

The high copper content of the weld metal in some of these older operating PWRs has reduced the fracture toughness properties of materials under neutron irradiation. (Staff's witness Litton, p. 4, following Tr. 13330; Applicant's witness Ranganath, p. 7, following Tr. 13280). However, those plants which meet the requirements of 10 CFR Appendix G with respect to fracture toughness do not fall within the concerns described by the Staff in generic issue A-11 (Staff's witness Litton, p. 4, following Tr. 13330).

342 The ACNGS pressure vessel has been fabricated to meet the requirements of Appendix G to Part 50 (Applicant's witness Ranganath, pp. 3-4, following Tr. 13280; Staff's witness Litton, p. 3, following Tr. 13330) as well as the surveillance requirements of 10 CFR Appendix H (Applicant's witness Ranganath, p. 5 following Tr. 13280; Staff's witness Litton, p. 3, following Tr. 13330). Since the ACNGS pressure vessel meets all the criteria of Appendices G and H to Part 50, has restrictions in the copper and phosphorus content of the vessel beltline material, and conservatively accounts for radiation effects set forth in Reg. Guide 1.99, the concerns of generic task A-11 do not apply to ACNGS. (Applicant's witness Ranganath, p. 8, following Tr.

13280; Staff's witness Litton, p. 4, following Tr. 13330). Accordingly, the Board finds that TexPirg's contention is without merit.

TexPirg Additional Contention 41: Reactor Pressure Limit/ Relief Valves

343 TexPirg alleges that there is inadequate protection against overpressurization of the ACNGS Reactor Coolant Pressure Boundary (RCPB) resulting from pressure increase transients. This concern arises, according to TexPirg, because the Nuclear Pressure Relief System (NPRS) is not designed adequately to ensure that during the most severe abnormal operational pressure increase transient, pressure is maintained below the limit allowed by the ASME Boiler & Pressure Vessel Code. TexPirg asserts that reliance on the high flux signal as a major contributor to the termination of the pressure transient does not provide an adequate assurance against overpressurization of the RCPB because there is a history of reportable occurences which indicate poor performance in the BWR flux instrumentation systems with inaccuracies of 5.4%.

344 The NPRS consists of 19 safety/relief valves located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. These valves protect against overpressurization of the RCPB by opening automatically upon receipt of pressure signals (relief operation) to limit a pressure rise or by self-actuation (safety operation), if not already automatically opened for relief operation. In order to protect against overpressurization of the reactor vessel the SRV setpoints are such that the peak vessel pressure does not exceed 110% of the design during the limiting pressurization event. (Applicant's witness Huang, pp. 8-9, following Tr. 16146).

345 The pressure transient resulting from the closures of all main steam line isolation valves (MSIV) represents the most severe pressurization transient when credit is taken only for an indirectly derived SCRAM. Normally, MSIV closure will cause an automatic SCRAM which is initiated through the Reactor Protection System (RPS). The RPS is provided with a direct SCRAM signal due to MSIV closure. The ASME Code allows for consideration of the immediate scram generated by MSIV closure. However, GE has performed an analysis of the MSIV pressure transient assuming the failure of the direct, safety-grade main

steam isolation valve position SCRAM. In this event the reactor is shutdown by the backup, indirect high-neutron flux SCRAM. This very conservative analysis introduces a significant delay between the initiation of the transient (MSIV closure) and the initiation of a SCRAM. Even under this unlikely scenario, the most severe over-pressure is terminated well below the pressure limit required by the ASME Code. (Applicant's witness Huang, pp. 10-12, following Tr. 16146).

346 There are four divisions of Average Power Range Monitors (APRMs) which measure neutron flux and thus reactor power levels. (Applicant's witness Huang, Tr. 16226). The reportable occurrences cited by TexPirg in its contention do not demonstrate a lack of reliability of these instruments to provide dependable reactor trips. This is because most reportable occurrences regarding these instruments result from setpoint drift exceeding allowable drift (Staff's witness Hodges, p. 13, following Tr. 15128). However, the APRM SCRAM setpoint assumed in the overpressure protection analysis is increased over the normal setpoint used in the plant to account for measurement uncertainty and drift. (Staff's witness Hodges, p. 13, following Tr. 15128; Applicant's witness

Huang, Tr. 16226).*/ Since the flux signal provides a reactor trip when the signal reaches a nominal 120% rated flux (Staff's witness Hodges, p. 12 following Tr. 15128) and since an overpressurization transient event such as MSIV closure results in a flux spike peak at approximately 300% of nominal rated flux (Id. at p. 13), set point drift of a few percent is insignificent when compared with the flux spike resulting from such a transient. The resulting effect on peak vessel pressure is still well below the ASME Code limits. (Applicant's witness Huang, p. 13 following Tr. 16146, Staff's witness Hodges, Tr. 17618-21, 17624-27).

347 The Board concludes that the pressure limits will not be exceeded even if the high-neutron flux SCRAM (the second or delayed SCRAM) is assumed to fail. The SCRAM under these conditions is initiated by the high reactor pressure trip signal. The probability of the simultaneous failure of the MSIV position SCRAM and high-neutron flux SCRAM signals is extremely low. But even assuming these extremely unlikely events, the peak reactor vessel pressure for this transient is still below the ASME Code

*/ Typically, the assumed setpoint in the analysis is 4% above the nominal setpoint (Applicant's witness Huang, p. 13 following Tr. 15128).

limit of 1,375 psig. (Applicant's witness Huang, p. 14, following Tr. 16146; Tr. 16223; Staff's witness Hodges, Tr. 17611-14). Hence, even if TexPirg's claims of high-flux signal unreliability were true, which they are not. the RCPB would still be adequately protected against overpressurization.

TexPirg Additional Contention 52: Outside Containment Sampling

348 TexPirg contends that as a result of the TMI incident Applicant should be required to have a system that permits taking of a primary coolant sample when the containment building is contaminated with radioactivity.

349 In March, 1981, the NRC Staff published NUREG-0718, "Licensing Requirements For Pending Applicants For Construction Permits And Manufacturing License." These requirements have been imposed as a result of the TMI incident. Section II.B.3 of NUREG-0718 requires that all construction permit applicants must review their sampling system design and commit to the necessary modifications for installation of a post-accident sampling system. Such a system should provide the capability to promptly obtain inside reactor coolant samples and

perform chemical and radionuclide analyses. Sufficient shielding must be installed to ensure that no person involved in sample taking or analysis will receive a radiation exposure in excess of General Design Criterion 19 (5 REM to the whole body or 75 REM to extremities). Additionally, NUREG-0718 refers the applicant to detailed requirements of NUREG-0737 (Clarification Of TMI Action Plan Requirements), which specifies that procedures be prepared which can relate the radionuclide analysis to the severity of core damage. (Staff's witness McCracken, p. 2; following Tr. 20078; Tr. 20088).

350 As indicated in Appendix O, page O-79 to the PSAR (App. Exh. 27), ACNGS will have the capability to collect liquid samples from the reactor coolant and suppression pool from outside the containment building. As it is now planned, the sampling system will be a manual system that can be operated to collect a liquid sample. The system will be capable of taking samples from several separate locations. The sampling station will be located in the Reactor Auxiliary Building near the Reactor Shield Building wall, in order to make the sample lines as short as possible. Shielding will be provided to ensure that doses to those personnel collecting the samples are maintained within GDC 19 limits as specified

in NUREG-0737. (Applicant's witness Robertson, pp. 12-13, following Tr. 15783; Staff's witness McCracken, pp. 2-3, following Tr. 20078; Tr. 20096).

351 The Board finds that the Applicant has provided assurance that the capability will exist for sampling and analyzing the reactor coolant at ACNGS following an accident in which there is core degradation, and that the Applicant will have the ability to relate the analysis to core damage. (Staff's witness McCracken, p.3, following Tr. 20078; Tr. 20090-97).

TexPirg Additional Contention 53: Non-Condensible Gases

352 TexPirg contends that ACNGS should be designed so that Applicant can determine how much non-condensible gas is in the reactor vessel. According to TexPirg, the non-condensible gas in question is hydrogen. (Tr. 17660-61). TexPirg has a concern, based on alleged occurrences during the Three Mile Island (TMI) incident, that hydrogen could build up in the reactor and create an explosion hazard.*/

*/ TexPirg's contention talks about monitoring "non-condensible gas . . . during an ECCS." As a result of information obtained during discovery, Applicant interpreted the contention to be concerned with a small break LOCA. (Applicant's witness Elliott, p. 3, following Tr. 15783). This is the only interpretation which makes sense because hydrogen would obviously vent out through a large break.

353 TexPirg failed to present any evidence that there could ever be an explosive build-up of hydrogen in the reactor at ACNGS. In contrast, Applicant and Staff clearly established that there will be insufficient oxygen in the ACNGS reactor vessel to support hydrogen combustion regardless of the amount of hydrogen in the reactor vessel. For events such as a small break LOCA the oxygen is generated primarily by radiolysis, i.e., the decomposition of water into hydrogen and oxygen caused by high energy radiation. Where there is an excess of hydrogen, such as the event postulated by TexPirg, the recombination reaction is faster than the decomposition reaction. Therefore, the free oxygen concentration would not increase and would not reach either the flammability limit of 4% or the detonation limit of 9%. (Applicant's witness Elliott, pp. 4-5, following Tr. 15783; Staff's witness Hodges, Tr. 17719-26; 17741-44).

354 Aside from the fact that there will not be a significant hydrogen buildup within the reactor vessel, the potential for hydrogen formation can be monitored by use of water level indicators. In order to generate a large scale metal-water reaction which produces hydrogen the reactor core must become uncovered. The reactor water level instrument at ACNGS gives a very reliable reading of the

reactor water level. As soon as the operator is warned of a problem with water level his immediate response is to ensure the water level gets back to normal following well defined emergency procedure guidelines. Accordingly, the fuel will stay covered and there will be no excess hydrogen generation during a small break LOCA. (Applicant's witness Elliott, pp. 3-4, following Tr. 15783; Tr. 15802-05; 15838-42; Staff's witness Hodges, Tr. 17698, 17729, 17739-41).

355 In the unlikely event that excess hydrogen were generated inside the reactor, it would be vented automatically through the normal pressure control devices. Where there is a small break and pressure remains high, non-condensible gases such as hydrogen would be vented through the safety relief valves and the RCIC steam turbine exhaust. Venting is also possible through the reactor head vent lines and associated valves. (Applicant's witness Elliott, pp. 5-7, following Tr. 15783; Staff's witness Hodges, p. 15, following Tr. 15128; Tr. 17678-79; 17735-39). In addition, if the break is large enough to depressurize the reactor, venting would occur through the break. (Staff's witness Hodges, p. 15, following Tr. 15128).

356 Unlike ACNGS, the design of TMI did not allow the operator to get a direct measure of the reactor water level nor could the TMI reactor be rapidly depressurized and/or vented. (Applicant's witness Elliott, p. 4, following Tr. 15783; Staff's witness Hodges, p. 15, following Tr. 15128; Tr. 17672-74).*/

357 Based on the evidence in this record, the Board finds that there is no merit to TexPirg's contention. There is no evidence that hydrogen could ever reach explosive levels within the reactor and thus there is no reason to monitor hydrogen within the reactor. Moreover, the water-level indicators provide an adequate indication of the potential for hydrogen formation within the reactor.

TexPirg Additional Contention 55: Rapid Depressurization/ Steam Break

358 TexPirg contends that in the event of a steam line break the reactor vessel will rapidly depressurize causing the steam bubbles to "froth" and thereby draw

---

*/ The reliability of the reactor water level indicators is the subject of Doherty Contention No. 41, supra.

coolant into the reactor. TexPirg alleges that this movement of water will cause a dangerous increase in reactivity before the reactor SCRAMS. (Tr. 15370). TexPirg's pleadings indicate that the source of this concern is derived from the SPERT tests conducted in June, 1970, by the Idaho Nuclear Experimental Laboratories as reported in IN-1370.*/

359 The statements in IN-1370 which serve as the basis for TexPirg's contention were untested and essentially conjecture at the time. Moreover, TexPirg has presented no evidence to substantiate the allegations in the contention. In contrast, GE has developed a detailed computer code which calculates among other things, the physical state of the reactor coolant after a large pipe rupture. The GE code is used to analyze the short-term thermodynamic and thermal-hydraulic behavior of the coolant in the vessel during a postulated loss-of-coolant accident (LOCA). In particular, the code predicts the core flow, core inlet enthalpy, and core pressure during the early stages of the reactor vessel

---

*/ "TexPirg Amendments and Replies to Applicant and Staff Contentions [sic] Submitted Pursuant to ALAB-535," June 15, 1979, p. 4.

blowdown. The code has been verified by GE in tests performed in GE's Two Loop Test Apparatus. (Applicant's witness Hamon, pp. 3-4, following Tr. 15721). GE has demonstrated that the actual consequences of rapid depressurization are exactly the opposite of what TexPirg alleges.

360 Under normal operating conditions, coolant water flowing into a BWR core is at a temperature and pressure close to the saturation point. After a guillotine break in the main steam line, pressure in the reactor core will fall rapidly. The depressurization rate averages 20 psi per second for the first 30 seconds after the break occurs and slowly decreases thereafter. Since the pressure at the break will be significantly lower (initially atmospheric pressure) than the water and steam inside the vessel, the water and steam in the rest of the vessel will move toward the break. Simultaneously, the rapid depressurization of the reactor vessel, caused by the escape of steam out of the break, will cause the water in the core to flash to steam rapidly. The rapid change into steam drastically decreases the effectiveness of the coolant as a moderator and, therefore, introduces a large amount of negative reactivity. Although the void generation in the core may briefly increase the volume of the moderator,

the creation of large voids will significantly reduce moderator density resulting in a large negative reactivity insertion. (Applicant's witness Hamon, pp. 2-5, following Tr. 15721; Tr. 15730-31; Staff's witness Hodges, p. 16, following Tr. 15128; Tr. 17754-58).

361 The Board finds that, contrary to TexPirg's contention, there is no mechanism for drawing cold water into the core quickly enough to adversely affect reactivity prior to full SCRAM.*/

## IV. BOARD QUESTIONS

On July 31, 1980, the Board issued an order listing the questions which the Board wished to have answered during the course of the hearings. With one exception, questions raised by the Board during the course of the hearings were related to the contentions being litigated and those questions have been addressed as part of the discussion of the individual contentions.*/ Accordingly, the following discussion relates only to the questions raised in the July 31 Order. As will be seen, a number of these questions were also addressed in

---

*/ Following SCRAM, injection of cold water is of no consequence as shown in the findings on Doherty Contention 7, LPCI Cold Slug, *supra*.

*/ The one exception related to questions on generic issues discussed below.

connection with related contentions. The Board has determined that each of its question has been answered satisfactorily.

1. Do the availability of lignite and the environmental costs of its use justify its consideration as an alternative fuel?

362 This question was answered by Applicant and Staff as part of their testimony on Doggett Contention 1(b) and will not be addressed in further detail here.

2. Did the Staff use WASH-1400 in arriving at its conclusions regarding environmental risks, as assessed in §S.7 of the FSFES?

363 Mr. Moon the NRC Staff Project Manager for Allens Creek, testified that the Staff did not rely on WASH-1400 in arriving at its conclusions regarding the environmental risks assessed in Section 5.7 of the FSFES (Staff Exh. 23). (Staff's witness Moon, following Tr. 21126).

3. In our Order of February 8, 1979, at page 24, we stated our desire that the Applicant testify on the record as to its intent and willingness to comply with NRC requirements regarding the subject of anticipated transients without scram.

364 This question was addressed by Applicant in connection with its testimony on Doherty Contention 8. Mr. Robertson, HL&P's Manager of Nuclear Licensing, testified that HL&P committed to incorporate any design modifications that may be required by the Staff to resolve the ATWS issue. (Applicant's witness Robertson, p. 3, following Tr. 15526). This commitment is noted at page C-3 of Supplement No. 4

to the ACNGS Safety Evaluation Report (Staff Exh. 21). More importantly, Applicant will have to comply with the ATWS rule to be adopted by the Commission. See, 46 Fed. Reg. 57521 (Nov. 24, 1981).

4A. Will the ACNGS facility meet current NRC requirements regarding standards for combustible gas control?

365 At the time this question was asked the applicable regulations governing control of combustible gas were found in 10 CFR § 50.44. Those regulations are still in effect and Applicant has demonstrated compliance. Applicant's PSAR thoroughly describes the hydrogen control system, and the Staff has reported on its review of the design in Section 6.2.5 of Supplement 2 to the Allens Creek SER (Staff Exh. 19). ACNGS will have the capability to sample and measure the hydrogen concentration throughout the drywell and containment during an accident. A drywell mixing system, combined with the thermal gradients existing during an accident, will ensure that there is an adequately mixed atmosphere in the containment and drywell following a postulated LOCA. Combustible gas concentrations in containment following a LOCA are controlled with an electric hydrogen recombiner system, which operates on the principle of natural convection of warm air rising through the recombiner.

As a secondary or back-up system to the hydrogen recombiners, a hydrogen purge system operating in conjunction with the Standby Gas Treatment System is available in the ACNGS design as an additional containment hydrogen control means. The hydrogen recombiner system is designed to maintain the amount of hydrogen in the containment below the 4% lower flammability limit. (Applicant's witness Weingart, pp. 9-11, following Tr. 15783; Staff's witness Fields, pp. 1-2, following Tr. 16267).*/

366 Subsequent to the issuance of the July 31 Order, the NRC promulgated additional requirements pertaining to combustible gas control as a result of the TMI incident. NUREG-0718, Item II.B.8, parts (3), (4)b and (4)c require that a hydrogen control system be provided to control the amount of hydrogen generated by a 100% active fuel-clad metal-water reaction that has been postulated to occur from a degraded core accident. HL&P has committed to provide such a hydrogen control system in the ACNGS design to meet this requirement. Currently a number of different methods are being considered throughout the industry and it is expected that these efforts will produce valuable data upon which to select an optimum means

---

*/ Also see the findings on TexPirg Additional Contention 34, Hydrogen Monitoring, supra.

of hydrogen control. Further, it is expected that the pending rulemaking on degraded cores will determine the necessity for such a system. (Applicant's witness Robertson, pp. 11-12, following Tr. 15783).

367 Supplement No. 3 to the Allens Creek SER (Staff Exh. 20) contains the Staff's review of Applicant's compliance with NUREG-0718, which has been adopted by the Commission as the so-called TMI requirements that apply to all plants at the construction permit stage of licensing. Supplement No. 3 describes the preliminary design as a postaccident inerting system (PAIS) using carbon dioxide ($CO_2$) as the system for hydrogen control. The Staff has concluded that the PAIS has the potential to become a suitable long-term solution to hydrogen management, but has required the Applicant to assess within 2 years after issuance of a construction permit, the value and costs of alternative hydrogen mitigation systems which may provide enhanced margins of safety relative to the PAIS. (Staff Exh. 20, p. II-8; Staff's witness Fields, pp. 2-3, following Tr. 16267).

368 During the evidentiary hearings, the Board raised a question as to whether there were any problems associated with inadvertent operation of the $CO_2$ inerting system, particularly with overpressurization of the containment.

Mr. Fields testified that following a complete metal-water reaction and inerting with $CO_2$, the containment pressure would reach 42 psig. Inadvertent operation of the $CO_2$ system would only pressurize the containment to 25 psig. The required minimum internal pressure corresponding to Service Level C stress limits for the design of steel containments such as the Mark III is 45 psig. Since ACNGS is designed to comply with these limits, there would be no catastrophic failure of the containment following inerting. (Staff's witness Fields, Tr. 16283-84; 16288-91, Applicant's witness Lugo, p. 3, following Tr. 19204).

4B. Will the ACNGS facility meet current NRC requirements with respect to GDC 50-Containment Design Basis?

369 General Design Criterion 50 requires that the reactor containment structure and associated support systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. The design margin must reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions such as required by § 50.44 (energy from metal-water) and other chemical reactions

that may result from degraded emergency core cooling functioning; (2) the limited experience and data available and (3) the conservatisms in the calculations. As stated in PSAR Section 3.1.2.5.1.1 (App. Exh. 27), HL&P is committed to meet the requirements of General Design Criterion 50. There is no problem in meeting this commitment because the design temperatures and pressures set forth in PSAR Section 3.8.2 exceed the pressures and temperatures associated with accidents described and evaluated in Chapters 6 and 15 of the PSAR. (Applicant's witness Cheng, pp. 7-8, following Tr. 15783).

370 The Staff agrees that the Applicant complies with the GDC 50 requirements (Staff's witness Fields, p. 2, following Tr. 12514 . The spectrum of accidents that the Applicant has analyzed in Chapters 6 and 15 of the PSAR (App. Exh. 27) include a variety of pipe break sizes and single active failures. The most conservative assumption regarding degraded emergency core cooling is loss of offsite power and loss of one of the two onsite emergency power supplies. This assumption, which has been analyzed by the Applicant, is more conservative than the loss of one ECCS line due to a break in that line because one entire ECCS train is lost when only one onsite power supply is available. (Id.). Also, for

Mark III containments, neither the short-term nor the long-term pressure and temperature response of the drywell and containment is affected by degraded emergency core cooling. This is because the short-term response is controlled by the first few seconds of the blowdown and the long-term response is controlled by the amount of energy that can be removed from the suppression pool by one train of the RHR heat removal system. (Staff's witness Fields, p. 2, following Tr. 12514; Applicant's witness Pappone, pp. 1-3, following Tr. 12491).

5. Should control rods, control rod drives and their hydraulic control units (listed as Safety Class 2 in Table 3.9-4 of the PSAR) be treated as Seismic Category 1 components in accordance with Reg. Guide 1.29?

371 Regulatory Guide 1.29 requires that reactor vessel internals and reactivity control systems, such as control rods and control rod drives, be designated as Seismic Category 1 and be designed to withstand the effects of a safe shutdown earthquake. All portions of the Control Rod Drive System necessary to shutdown the reactor are classified as Seismic Category I, including control rods, control rod drives and hydraulic control units. (Applicant's witness Ross, pp. 1-3, following Tr. 12690; Staff's witness Leung, p. 2, following Tr. 12992).

6. In our Order of March 10, 1980, ruling upon contentions, we rejected Mr. Doherty's Contention 38C but noted that he had referred (during the October 15-19, 1979 Prehearing Conference) to two ACRS letters to the NRC expressing concerns about the design of reactor heat removal systems (Tr. 1124). The Board is interested in these concerns of the ACRS and requests the Staff and/or Applicant to present testimony as to whether these concerns are applicable to the ACNGS and, if so, what remedial measures are planned.

372 The letters referenced in the question raise two separate concerns. First, in a letter to Dixy Lee Ray from W. R. Stratton, December 12, 1974, the ACRS expressed concern about the heat removal capability of the RHR system assuming a single failure in the let-down line for the shutdown cooling mode of operation. Second, in a letter from Dade Moeller to Marcus Rowden, December 1976, the ACRS expressed concern over the possibility of damage to a heat exchanger of the residual heat removal system by overpressurization or by hydrodynamic forces that could conceivably result from valve malfunction. This is associated with the steam condensing mode of operation of the RHR when the reactor core isolation cooling system is in use. (Staff witness Hodges, p. 17, following Tr. 15128).

As to the shutdown cooling mode of operation, General Design Criterion 34 requires that the capability exists to remove residual heat from the reactor core using only safety grade equipment and considering a single active failure. For the shutdown cooling mode of operation, the RHR system draws water from a single pipe which is connected to a recirculation loop. The concern expressed by the ACRS was that failure to open a single valve in that pipe could incapacitate the shutdown cooling mode of operation. However, it has been determined that shutdown cooling can also be achieved by flooding the vessel to the elevation of the steamlines, opening several of the automatic depressurization system (ADS) valves to discharge heated water to the suppression pool, and pumping water back into the vessel from the suppression pool through the RHR heat exchangers. The RHR system operating in the low pressure injection mode would thus remove the residual heat discharged to the suppression pool via the ADS valves. (Staff's witness Hodges, pp. 17-18, following Tr. 15128; Tr. 17807-12; 17834-37).

373 As to the second question, the RHR heat exchangers are of the shell and tube design. Primary water flows through the shell side (outside of tubes in a tube bundle) and cooling water flows inside the tubes. For the steam condensing mode of operation, water is drained from the shell side until the shell side level is about 75% of the level set point. Steam is admitted at the top

of the shell and is condensed on the outer surfaces of the tubes in the tube bundle. The ACRS was concerned that if cold water were admitted to the shell side of the heat exchanger (as a result of a valve failure or operator error) during this mode of operation, then rapid condensation of the steam might result in water hammer in the heat exchanger. The water hammer could result in overpressurization of the heat exchanger and lead to unacceptable hydrodynamic loads on the heat exchanger and its supports. The fact is that water-hammer has never been observed in the steam condensing mode. While the Staff has undertaken a generic study of water hammer (See Unresolved Safety Issue A-1, Water-hammer),*/ RHR system water hammer has never resulted in a loss of RHR function. As part of the generic evaluation of water hammer, the Staff is developing procedures to minimize the potential for water hammer. The operational steps that have been taken in the interim are adequate to protect the public health and safety because water hammer damage to an RHR system has never resulted in an unsafe condition. (Staff's witness Hodges, pp. 18-19, following Tr. 15128; Tr. 17840-48; 17851).

---

*/ Also, see the findings on Doherty Contention No. 44, IGSCC/ Water Hammer, supra.

7. In our Order of March 10, 1980, ruling upon contentions, we requested, at page 34, that the Staff present evidence in response to the following Board questions:

> Is there an opportunity for the permissible site boundary radiation level to be exceeded by virtue of a gap in NRC and/or EPA regulations, whereby an on-site transportation accident gives rise to a radiation field which, when added to the ambient radiation level from normal plant operation (including radiation from stored spent fuel), might then result in a higher than permissible site boundary radiation level? If not, why not? If so, does this constitute an oversight in the Staff's FES analysis?

374 Mr. Moon, the Staff's Licensing Project Manager for ACNGS, testified that the Staff had found nothing to indicate that there should be a concern with an on-site transportation accident. The design criteria for shipping containers are sufficient to permit transportation over public highways, and an accident is less likely to happen on site than on a public highway. Thus, this on-site accident does not have significant enough radiological consequences to warrant an accident analysis, (Staff's witness Moon, p. 2, following Tr. 21135, Tr. 21137).

8. The Board requests the Applicant and/or Staff to present evidence regarding the adequacy of the ACNGS reactor building's foundation design with respect to the ability of subsurface soil to support such a heavy structure. Soil mechanics rather than subsidence is of concern here, with respect to avoiding unacceptable settling of heavy structures. This item appears not to have been addressed in our Partial Initial Decisions.

375 The basic formations of the subsurface soil materials at the ACNGS site have been identified as Beaumont, Montgomery, Goliad, and Fleming formations. Undisturbed soil samples were extracted from each of these formations and subjected to laboratory static and dynamic tests to investigate their strength characteristics, compressibility under heavy load and dynamic properties. The results of the laboratory tests, along with engineering interpretations, are presented in PSAR Section 2.5.6 (App. Exh. 27). The results show that the soils at this site have high shear strengths and low compressibility. This data confirms that ACNGS is founded on soils which are more than capable of sustaining the loads to be imposed.

376 The reactor building mat foundation will rest on the Montgomery formation, which is a very dense and highly compact granular sand material. The maximum allowable bearing pressure for the reactor mat foundation design is 10 kips per square foot (ksf) under the static loading conditions. The mat foundation has a safety factor greater than 20. Normally a safety factor of 1.5 is considered acceptable. Accordingly, the soil can clearly support the reactor building. (Applicant's witness Mercurio, pp. 2-5, following Tr. 16656).

9. In our Order of March 10, 1980, ruling upon contentions, we requested, at page 24, that the Staff provide an evidentiary response to the following question:

> What is the technical basis for concluding that maintaining containment atmosphere temperature and relative humidity values within prescribed limits is a practical method for minimizing bypass leakage?

377 Containment atmosphere relative humidity and temperature limits are not used to control bypass leakage, but are concerned with the sizing analysis for the containment vacuum breaker system. The Staff's basis for this sizing analysis is set forth in Section 6.2.1(2) of SER Supplement No. 2 (Staff Exh. 19). (Staff's witness Fields, p. 3, following Tr. 12514).

10. Applicant is requested to verify during the evidentiary portion of this proceeding whether the ACNGS drywell will be tested at some prespecified value in excess of design pressure.

378 Applicant states in PSAR Section 3.8.3.7 (App. Exh. 27) that the ACNGS drywell will be tested at 34.5 psig. This is equivalent to 115% of design pressure (30 psig) in accordance with NRC Regulatory Guide 1.18 recommendations for concrete containments. (Applicant's witness Cheng, p. 2, following Tr. 16111).

11. Applicant's Environmental Report (Amendment 0, November 13, 1973, p. 5.6-2A) states in part that ". . . no instance of significant bird losses have been reported." Since we do not find this matter in the FES as supplemented, we request the Staff to present evidence upon this matter.

379 This question was answered by Applicant and Staff as part of their testimony on Marrack Contention 2(c) and need not be addressed in further detail here.

12. In our Order of March 10, 1980, ruling upon contentions we requested, at page 50, that the Applicant address the following question during the evidentiary phase of this proceeding:

> Has it been definitively established whether a liquefied petroleum gas pipeline, located in the vicinity of the proposed plant site, might carry potentially more dangerous materials such that, following a pipeline rupture, safe shutdown of the plant could be precluded?

380 This question was addressed by Applicant and Staff as part of their testimony on Bishop Contention 6 and need not be addressed in further detail here.

13. Under date of March 29, 1980, a letter from Mr. Clarence Johnson of PIRG to this Board advised that the output from Applicant's proposed two lignite fired plants north of Houston, accounted for in the Staff's FSFES at Tables S.8.13 and S.8.14, should now be reassessed because a prospective loadsharing agreement between Applicant and the Dow Chemical Company did not materialize. We request the Staff to clarify this matter during the evidentiary phase of the proceeding.

381 The Applicant's proposed capacity additions were considered in connection with Applicant's testimony on need for power and on Doggett Contention 1(b), TexPirg Additional Contention 8 and TexPirg Contention 7. The question need not be discussed in further detail here.

14. In our Order of April 10, 1980, we requested that Applicant and Staff present evidence in response to the following Board question:

If there is no difference in principle between the ACNGS and the Dresden III MSLRM systems, why couldn't the Dresden III incident cited in the referenced Order, be repeated at the ACNGS?

382 This question was addressed in connection with the findings on Doherty Contention No. 14 and need not be discussed in further detail here.

15. There are numerous technical subjects that have arisen since the TMI-2 incident that appear to have applicability to the ACNGS. We request Applicant and Staff to be prepared to advise us at the prehearing conference beginning on August 13, 1980, how these matters will be handled at the evidentiary hearings.

383 NUREG-0660, "NRC Action Plan Developed as a Result of the TMI Accident" (May, 1980), was developed to provide a comprehensive and integrated plan for the actions judged appropriate by the NRC to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at TMI Unit 2 and the official studies and investigations of the accident. However, the TMI-2 Action Plan (NUREG-0660) did not specifically address the requirements to be applied at the construction permit stage of licensing. Therefore, the Staff developed a separate set of requirements for the construction permit plants. These requirements are contained in NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License" (March, 1981).

384 Applicant's response to the requirements of NUREG-0718 is provided in Amendments 57 and 59 to the PSAR. These amendments were reviewed by the Staff and the results of the review were reported in Supplement No. 3 to the Allens Creek SER, NUREG-0515 (July 1981) (Staff Exh. 20). The Staff has concluded that the information supplied by the Applicant in Amendments 57 and 59 to the PSAR is sufficient to demonstrate compliance with NUREG-0718.

16. In our Order of February 9, 1979, at page 7, we noted that the Staff should address unresolved generic safety issues during the construction permit hearing.

385 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 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. (Staff Exh. 19, SER Supp. 2, p. C-1).

386 In some cases, immediate action is taken to assure safety, and 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, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. Often further study is required before a judgment can be made as to whether existing NRC staff requirements should be modified to address the issue. These issues are sometimes called "generic safety issues" or "unresolved safety issues." 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 view is underway. (Id.)

387 The NRC has created a Technical Activities Steering Committee to increase high level management involvement in and to improve management oversight of generic technical activities. The Committee's functions include assigning proposed generic tasks to priority categories, assigning lead responsibility to a Nuclear Reactor Regulation (NRR) division for defining and executing each generic task, approving Task Action Plans, and regularly reviewing the progress of ongoing tasks. (Id.)

388 The generic issues that were considered by the Steering Committee included those from the Advisory Committee on Reactor Safeguard's listing, those listed in NRR's former Technical Safety Activities Report, the 27 issues discussed in "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff" (NUREG-0138) and "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from Director, NRR to NRR Staff" (NUREG-0153), and a number of other generic issues that were identified from a variety of different sources. Issues were assigned to various priority categories based on their judged safety, environmental or safety importance or their potential for improving the efficiency or effectiveness of the licensing process. (*Id.*)

389 Each of the generic issues applicable to ACNGS and their current status are thoroughly described in Appendix C of Supplement No. 2 to the SER (Staff Exh. 19), March, 1979.*/

---

*/ Appendix C also discusses the "unresolved safety issues" that have been identified as a result of Section 210 of the Energy Reorganization Act of 1974. Appendix D contains a listing of generic matters that have been raised by the ACRS. Both of these listings are cross-indexed to the Steering Committee's list of generic items and tasks.

This list was updated in Supplement No. 4 to the SER (Staff Exh. 21).

390 In response to additional questions from the Board on November 19, 1981 (Tr. 20415-16), the Staff explained that it considers an issue to be "technically resolved" at the time NRC management approves the issuance of a NUREG report that describes the Staff's conclusions of what requirements need to be implemented on operating plants and new plants to resolve the issue. Technical resolution involves incorporation of the Staff's conclusion into the NRC's Regulations, Standard Review Plan, and Regulatory Guides. As generic safety issues are resolved in this manner, they will be applied to ACNGS in the operating license review. (Staff's witness Moon, pp. 1-3, 6, following Tr. 21233). Mr. Moon provided a list of the issues which have been resolved, and testified that each would be incorporated into ACNGS on an individual implementation schedule (Staff's witness Moon, pp. 4-7 following Tr. 21233; Tr. 21278-79). As to those issues that have been resolved, the Staff is confident that their procedures produce resolutions that are satisfactory. (Staff's witness Moon, pp. 8-9 following Tr. 21233; Tr. 21281-82, 21286-89).

391 Based on the information contained in Staff Exhibits 19 and 21, and as further explained by Mr. Moon, the Board finds that the Staff has satisfied the requirements of Gulf States Utilities Company (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (Nov. 23, 1977).

17. In light of a Board Notification of November 26, 1979 (BN-79-41), the Staff shall present evidence as to the acceptability of using non-safety grade equipment for the mitigation of transients.

392 The concern discussed in BN-79-41 arose from a problem that related solely to Westinghouse pressurized water reactors. Westinghouse had determined that certain non-safety grade equipment with protective functions could be damaged as a result of a high energy line break inside or outside of containment. However, the Board is satisfied that this is not a problem for ACNGS or any BWR.

393 In a BWR-6, breaks outside containment, except for some instrument line breaks and a break in the scram discharge system, are isolated automatically. Except for the effects of direct impingement, the environment resulting from the break will not cause failure of control equipment. As to the noted exceptions, the instrument line break has been analyzed and found to result in acceptable consequences. The probability of a break in the scram

discharge volume is considered to be extremely low; however, the operator has sufficient time and information to depressurize and thus reduce the effect of the break. Additionally, for Mark III containments, such as ACNGS, the scram discharge system is located inside the containment.

394 The principal non-safety grade equipment inside containment consists of the recirculation pumps, relief valve actuators and instrument lines. Failure of the recirculation pumps would have minimal impact on the transient resulting from a high energy line break inside containment; in fact, power is assumed to be lost to these pumps in analyses of breaks inside containment. Spurious opening of a relief valve would not be detrimental, just a nuisance. Finally, instrument lines would not heat up quickly enough to adversely affect safety system actuation. (Staff witness Hodges, pp. 20-21, following Tr. 15128; Tr. 17857-66.)

## V. CONCLUSIONS OF LAW

395 The Board has given careful consideration to all of the documentary and oral evidence presented by the parties. Based on our review of the entire record in this proceeding and the foregoing findings and the findings set forth in our Partial Initial Decision (2 NRC 776) and

in accordance with the Notice of Hearing issued in the proceeding and the Commission's regulations, the Board concludes as follows:

A. The Application and the record of the proceeding, contain sufficient information, and the review of the Application by the Staff has been adequate to support the foregoing findings and following Conclusions and Order:

B. In accordance with 10 CFR 50.35(a):

(1) The Applicant has described the proposed design of the facility, including but not limited to principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.

(2) Such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report.

(3) Safety features or components, if any, which require research and development have been described by the Applicant, and the Applicant has identified, and there will be conducted, a research

and development program designed to resolve any safety questions associated with such features or components.

(4) On the basis of the foregoing, there is reasonable assurance that (a) such questions will be satisfactorily resolved before completion of the facility; (b) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

C. The Applicant is technically qualified to design and construct the proposed facility.

D. The Applicant is financially qualified to design and construct the proposed facility.

E. The issuance of a permit for construction of the facility would not be inimical to the common defense and security or to health and safety of the public.

F. The Board has independently considered and decided all matters in controversy among the parties and determines that the appropriate action to be taken is issuance of the construction permit for the facility with the conditions set forth on pages 1-2 to 1-3 of SER, Supp. 4 (Staff Exh. 21).

G. There is reasonable assurance that the site for ACNGS Unit 1 is a suitable location of a nuclear power reactor of the general type and size proposed from the standpoint of radiological health and safety considerations under the Act and the rules and regulations promulgated by the Commission pursuant thereto.

## VI. ORDER

In accordance with 10 CFR §§ 2.760, 2.762, 2.764, 2.785, 2.786, this Initial Decision shall become effective immediately and shall constitute, with respect to the matters covered therein, the final action of the NRC forty-five (45) days after the date of issuance hereof, subject to any review pursuant to the provisions of 10 CFR §2.764(e). Exceptions to this Initial Decision may be filed by any party within ten (10) days after service of this Initial Decision. Within thirty (30) days thereafter (forty (40) days in the case of the NRC Staff) any party filing such exceptions shall file a brief in support thereof. Within thirty (30) days of the filing of the brief of the Appellant (forty (40) days in the case of the NRC Staff), any other party may file a brief in support of, or in opposition to, the exceptions.

IT IS SO ORDERED.

# APPENDIX A:

# LIST OF EXHIBITS

## EXHIBIT INDEX - APPLICANT

| Exhibit No. | Date | Page Identified | Page Received | Exhibit Description |
|---|---|---|---|---|
| 13 | 01/23/81 | 3851 | 3854 | Environmental Report Supplement (2 volumes) |
| 14 | 02/04/81 | 4535 | 4537 | Vickery/Gibson ACNGS Meeting Notes - Austin, Texas 09/06/77 |
| 15 | 03/05/81 | 8572 | 8610 | TERA Corp. Coastal Site Comparison Report |
| 16 | 03/17/81 | 9152 | 9197 | Dames & Moore Report: Environmental Assessment and Responses to NRC Questions - Transportation of RPV |
| 17 | 03/18/81 | 9345 | 9346 | ENR report re barge transport of RPV |
| 18 | 03/19/81 | 9578 | 9581 | Letter from McLelland (American Rigging) to McGuire (HL&P) of 10/28/80 re San Bernard River barge traffic. |
| 19 | 05/11/81 | 9899 | 9900 | Reference copy of STP Participation Agreement of 07/01/73 |
| 20 | 05/20/81 | 11457 | 11470 | Diagram of ACNGS cooling lake (from ER Supplement S2.1-2 (modified) |
| 21 | 06/05/81 | 13345 | 13345 | Affidavit of Guy Martin, Jr. - 05/11/81 |
| 22 | 06/05/81 | 13347 | 13347 | Letter from Goldberg (HL&P) to Denton (NRC) of 05/26/81 re HL&P's commitment to install jetty system to stabalize Brazos River bank near ACNGS |
| 23A | 06/08/81 | 13575 | 13577 | Diagram of fuel bundle cell |

## EXHIBIT INDEX - APPLICANT

| Exhibit No. | Date | Page Identified | Page Received | Exhibit Description |
|---|---|---|---|---|
| 23B | 06/08/81 | 13575 | 13577 | Diagram of fuel bundles |
| 24 | 06/08/81 | 13609 | ——— | Letter from Varner (HL&P) to Mason (GE) of 04/21/81 re NSSS Contract AC-2000 - new improved heat treated jet pump beams |
| | 08/17/81 | ——— | 14629 | |
| 24A | 08/17/81 | 14628 | 14629 | Letter from Leone (GE) to Horn (HL&P) of 01/19/81 re jet pump beams |
| 24B | 08/17/81 | 14630 | 14630 | Letter from Mason (GE) to Varner (HL&P) of 02/19/81 re NSSS Unit #1 - new improved heat treated jet pump beams |
| 25 | 10/08/81 | 18323 | 18324 | Letter from Seidle (HL&P) to Oprea (HL&P) of 10/28/80 enclosing STP inspection report dated 09/80 |
| 26 | 10/30/81 | 19855 | 19861 | Letter from Lee (HL&P) to Horn (HL&P) of 10/12/81 re Sheridan-Houston 6" LPG pipeline |
| 27 | 12/07/81 | 20807 | 20807 | PSAR (27 Volumes) |
| 28 | 12/07/81 | 20808 | 20808 | Amended Application |

DB06:D

## EXHIBIT INDEX - STAFF

| Exhibit No. | Date | Page Identified | Page Received | Exhibit Description |
|---|---|---|---|---|
| 12 | 01/23/81 | 3871 | 3871 | Final Supplement to Final Environmental Statement NGS - Unit 1, Aug. 1978 |
| 13 | 05/12/81 | 10109 | 10109 | Supplement No. 2 to AC FES |
| 14 | 06/01/81 | 12481 | 12484 | Letter from Sohinki to Board of 05/29/81 re Codell testimony concerning yearly dredging of San Bernard River |
| 15 | 06/10/81 | 13988 | 13989 | NRC report of 03/78 entitled "Fuel Failure Detection in Operating Reactors" |
| 16 | 10/30/81 | 19762 | 19767 | Affidavit of R. L. Gotchy - 01/25/78 |
| 17 | 12/07/81 | 20780 | 20780 | Letter from Sohinki to Board of 12/03/81 re Peterson testimony |
| 18, 19, 20, 21 | 12/09/81 | 21218 | 21226 | SER, Supplement Nos. 1, 2, 3, 4 |
| 22 | 12/09/81 | 21231 | 21231 | Affidavit from Director of Nuclear Reactor Regulations verifying authenticity and accuracy of Staff Exhibit 18 as Microfiche copy of SER Supplement No. 1 |
| 23 | 12/09/81 | 21246 | 21248 | Original SER of 11/74 |

DB06:E

EXHIBIT INDEX - INTERVENORS

| Intervenor Exhibit No. | Date | Page Identified | Page Received | Exhibit Description |
|---|---|---|---|---|
| Doherty-1 | 03/18/81 | 9362 | 9369 | Galveston Corps of Engineers Hydrographic Bulletin - Channels with project depths under 25 feet dated 01/01/81 |
| TexPirg-8 & 9 | 05/15/81 | 10837 | 10840 | 8 - "Permit to Appropriate State Water", Application No. 3229, Permit No. 3233, Permitte: HL&P Co., granted 02/24/76 on Colorado River<br><br>9 - "An Order granting Application No. 3229 of HL&P Co., et al", by Texas Water Rights Commission. |
| Doherty-2 | 05/21/81 | 11898 | 11906 | U.S. Atomic Energy Commission Report by Idaho Nuclear Corp. - R. W. Miller entitled "The Effects of Burnup On Fuel Failure dated 12/70 |
| TexPirg-15 | 06/03/81 | 12908 | 12925 | Galveston Corps of Engineers public notice entitled "Maintenance Dredging Gulf Intracoastal Waterway - San Bernard River, Texas" dated 12/13/74 |
| Doherty-3 | 06/10/81 | 14047 | 14050 | Nuclear Safety article entitled - "Assessment of Light - Water - Reactor Fuel Damage During a Reactivity - Initiated Accident dated 09-10/80 (Vol. 21, No. 5) |
| Doherty-3A | 06/10/81 | 14049 | 14050 | MacDonald/Martinson (EG&G Idaho, Inc.) report entitled "Response of Preirradiated Fuel Rod Bundle During Reactivity Initiated Accident Test 1-4" dated 10/27-10/31/80 |

## EXHIBIT INDEX - INTERVENORS

| Intervenor Exhibit No. | Date | Page Identified | Page Received | Exhibit Description |
|---|---|---|---|---|
| Doherty-4 | 10/08/81 | 18223 | 18360 | Article from Fed. Reg. Vol. 45, No. 92 p. 30753 dated 05/09/80 - HL&P Co. (STP 1 & 2) Order to Show Cause (Effective Immediately) |
| Doherty-5 | 10/08/81 | 18305 | 18325 | NUREG-0834 NRC document entitled "NRC Licensee Assessments" dated 08/81 |

DB06:F

APPENDIX B:

WITNESS INDEX

## WITNESS INDEX

## WITNESS INDEX

## WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Cheng | 08/25/81 | Board Question 4B-<br>Containment Design Basis | 15739-15883 |
| | 08/26/81 | Board Question 10-<br>Drywell Pressure Testing | 16108-16119 |
| Chiou | 10/29/81 | Doherty 11-<br>Fuel Pool Accident | 19502-19553 |
| Chopra | 08/18/81 | Doherty 10-Diesel<br>Generator Reliability,<br>Doherty Contention 30-<br>Interconnection/grid<br>stability | 14637-14697<br>14806-14882 |
| Clarke | 08/18/81 | Doherty 10-Diesel<br>Generator Reliability | 14698-14805 |
| Codell | 05/12/81 | Potential Requirements For<br>Dredging of San Bernard<br>River To Transport Reactor<br>Pressure Vessel | 10158-10199 |
| Congdon | 11/16/81 | Doherty 15-Wigle Code | 20024-20068 |
| Dean | 09/15/81 | Baker 1-Financial<br>Qualifications | 16712-16889 |
| Dick | 02/13/81,<br>02/25/81,<br>02/26/81 | TexPirg 5, 7(a-d),<br>AC 8, AC 12, Doggett 1(b)<br>Board Question 1,<br>Request 13-<br>Alternative Energy Sources | 6213-6297<br>7017-7173 |
| Dillman | 08/17/81 | Doherty 6-Recirculation<br>Pump Overspeed | 14528-14582 |
| Dunlap | 11/17/81 | Doherty 45-<br>Core Lateral Support | 20102-20174 |
| Eckert | 06/10/81 | Doherty 7-<br>LPCI Cold Slug | 13883-13974 |
| Edwards | 09/16/81 | Need For Power | 16897-17128 |

WITNESS INDEX

## WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Gordon | 11/16/81 | TexPirg 10-IGSCC; Doherty 43-Stainless Steel Cleaning; Doherty 44-IGSCC/Water Manner | 19878-19968 |
| Gotchy | 01/21/81, 02/02/81-02/03/81 | Bishop 12, 21-Cooling Lake Radioactivity | 3240-3468<br>3882-4233 |
| | 10/30/81 | Cumings 9-Health Effects of Low Level Radiation | 19652-19745 |
| | | Potential Public Health Impacts of RN-222 Releases Resulting From Uranium Mining and Milling | 19745-1768<br>19823-19824<br>19851-19853 |
| Griffin | 09/18/81 | Schuessler Consolidated 1-Emergency Planning | 17401-17567 |
| Grubb | 11/17/81 | Doherty 45-Core Lateral Support | 20175-20210 |
| Gunther | 10/09/81 | Doherty 35-Welding | 18514-18601 |
| | 10/30/81; 11/16/81 | TexPirg 10-IGSCC; Doherty 44-IGSCC/Water Hammer; Doherty 43-Stainless Steel Cleaning; Board Question-Reg. Guide 1.54 | 19825-19851<br>19878-19968 |
| Guy | 02/09/81-02/10/81 | Energy Alternatives | 5144-5385<br>5474-5512 |
| | 09/16/81 | Need for Power | 16897-17128 |
| Hamilton | 02/23/81, 02/24/81 | Doggett 1(b), Bishop 13-Health Effects of Coal and Nuclear | 6327-6690 |
| | 10/29/81-10/30/81 | Health Effects of Low Level Radiation | 19555-19651 |

## WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Hamon | 08/24/81 | TexPirg AC 55-Rapid Depressurization Steam Break | 15715-15732 |
| Harrison | 06/02/81 | TexPirg 12-Cable Fires | 12639-12681 |
| Hartzman | 05/19/81 | TexPirg AC 26-Computer Code Error | 11282-11309 |
| Heidt | 11/19/81 | Board Questions re TexPirg AC 38-SCRAM Discharge Volume Float Switch | 20607-20634 |
| Henderson | 03/18/81-03/19/81 | TexPirg AC 1 Barge Slip | 9289-9647 |
| Hinkle | 02/24/81 | TexPirg 5, 7(a-d), AC 8, AC 12, Doggett 1(b), Board Question 1, Request 13-Alternative Energy Sources | 6693-6780 |
| Hobson | 11/17/81 | Doherty 45-Core Lateral Support | 20102-20174 (20146-20151 in camera) |
| Hodges | 08/20/81, 08/21/81, 10/05/81-10/06/81, 12/07/81 | Doherty 8, 17, 32; TexPirg AC 41, AC 53, AC 55, Board Questions 6 & 17, Doherty 42-SRV Position Indicator | 15118-15421; 17605-17885; 20817-20898 |
| | 10/06/81 | Doherty 38B-Cold Shutdown in 24 Hours; Doherty 41-Reactor Water Level Indicators | 17913-17985 |
| Holtzclaw | 05/21/81-05/22/81 | Doherty 3-Fuel Specific Enthalpy, Doherty 39-Fuel Swelling, Doherty 20(a)-Gap Conductance | 11718-12228 |
| Horton | 06/08/81 | Doherty 25-Fuel Failure/Flow Blockage, Doherty 14-Main Steam Line Radiation Monitor | 13364-13607 |

WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Hou | 10/27/81-<br>10/28/81 | Doherty 28-<br>Control Rod Ejection | 19055-19096 |
| | 10/28/81 | TexPirg 11, Doherty 31-<br>Flow Induced Vibration | 19163-19182 |
| Huang | 08/26/81-<br>08/27/81 | TexPirg AC 41-<br>Reactor Pressure Limit/<br>Safety Relief Valves | 16123-16230 |
| | 10/06/81 | Doherty 41, TexPirg<br>AC 54-Reactor<br>Water Level Indicators | 17938-17985 |
| | 11/20/81 | Doherty 21-ODYN Code | 20754-20766 |
| Hucik | 11/18/81 | Doherty 5-<br>Suppression Pool Uplift;<br>TexPirg AC 34-<br>Hydrogen Monitoring | 20283-20404 |
| Hussey | 03/17/81-<br>03/18/81 | TexPirg AC 1-<br>Barge Slip | 9150-9647 |
| | 03/19/81 | TexPirg 1-<br>STP 3 vs. AC 1 | 9659-9786 |
| Iotti | 05/19/81-<br>05/20/81 | Bishop 4, 5, 7, 9, 10-<br>Pipeline Rupture | 11391-11642 |
| | 06/04/81 | Doherty 47-<br>Turbine Missiles | 13062-13144 |
| | 09/16/81-<br>09/17/81 | Bishop 6, Board<br>Question 12-<br>Liquid Petroleum<br>Gas Pipeline | 17131-17236 |
| | 09/17/81 | Bishop 17-<br>TNT Detonation | 17284-17301 |
| Johnson, C. | 05/15/81 | TexPirg AC 1-Alternative<br>Sites, TexPirg AC 8-<br>Natural Gas Generation<br>and TexPirg 7a-c, Doggett<br>1(a)-Energy Conservation | 10647-11026 |

## WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Johnson, K. | 02/25/81 | TexPirg 5, 7(a-d), AC 8, AC 12, Doggett 1(b), Board Question 1, Request 13-Alternative Energy Sources | 6830-7011 |
| Kantor | 10/26/81-10/27/81 | Schuessler Consolidated 1-Emergency Planning | 18616-18993 |
| Kennedy | 11/19/81 | Board Questions-Scram Discharge Level Monitoring System | 20639-10687 |
| Kerley | 02/25/81 | TexPirg 5, 7(a-d), AC 8, AC 12, Doggett 1(b), Board Question 1, Request 13 Alternative Energy Sources | 6789-6829 |
| Knight | 11/19/81 | Board Questions-Scram Discharge Level Monitoring System | 20639-20687 |
| Korde | 05/19/81 | TexPirg AC 26-Computer Code Error | 11239-11280 |
| Lane | 08/21/81 | Doherty 41- Reactor Water Level Indicators | 15422-15511 |
| Lawhn | 09/18/81 | Schuessler Consolidated 1-Emergency Planning | 17401-17567 |
| Lee | 06/01/81 | TexPirg AC 30, McCorkle 17-Charcoal Adsorber Files | 12461-12479 |
| Leung | 06/04/81 | ASLB Question 5, Doherty 48-Seismic Category I Control Rods and Control Rod Drive Return Line | 12986-13060 |
| | | Doherty 6-Recirculation Pump Overspeed | 13036-13060 |
| | 11/20/81 | Doherty 11-Spent Fuel Pool Accident | 20732-20754 |

## WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Litton | 06/04/81 | Doherty 47-Turbine Missiles | 13145-13174 |
| | 06/05/81 | Doherty 50- Jet Pump Hold-Down Beams | 13181-13230 |
| | 06/05/81 | TexPirg AC 39- Fracture Toughness | 13329-13343 |
| | 10/30/81 | Doherty 35-Welder Training | 19768-19823 |
| | 11/16/81 | TexPirg 10-IGSCC Doherty 44-IGSCC/ Water Hammer | 19969-20013 |
| | | Doherty 43-Stainless Steel Cleaning, Reg. Guide 1.54 | 20013-20024 |
| Lugo | 10/28/81 | TexPirg AC 6- Mannings Coefficient | 19183-19272 |
| | 10/29/81 | Board Question-Service Level Stress Limits | 19360-19369 |
| Malec | 10/29/81 | Board Question-Service Level Stree Limits | 19360-19369 |
| | | McCorkle 17-Bypass Leakage | 19457-19495 |
| | | Doherty 11- Fuel Pool Accident | 19502-19553 |
| | 10/30/81; 11/16/81 | TexPirg 10- IGSCC; Doherty 44- IGSCC/Water Hammer, Doherty 43- Stainless Steel Cleaning, Board Question- Reg. Guide 1.54 | 19825-19851; 19878-19968 |
| Marrack | 02/04/81- 02/05/81, 08/19/81 | Marrack 2 & 4- Cooling Lake/ Recreational Benefits | 4365-4496 4514-4698 14888-15100 |
| | 06/03/81 | TexPirg 1-STP vs. AC, TexPirg AC 1-Barge Slip | 12748-12962 |

WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Martin | 08/27/81 | McCorkle 9- Chlorine Monitoring | 16240-16256 |
| | 08/28/81 | Doherty 40- Part 100 Releases | 16325-16401 |
| | 10/29/81 | McCorkle 17- Bypass Leakage | 19458-19495 |
| | — | Doherty 11- Fuel Pool Accident | 19502-19553 |
| Maurer | 12/09/81 | Board Question-Dredging of San Bernard Channel | 21160-21215 |
| McCracken | 11/17/81 | TexPirg AC 52-Outside Containment Sampling | 20073-20101 |
| McCuistion | 05/11/81-05/12/81 | TexPirg 1-STP 3 vs. AC 1 | 9855-10106 |
| McGuire | 01/23/81 | Environmental Report Supplement | 3852-3855 |
| McIntyre | 10/28/81 | TexPirg AC 6- Mannings Coefficient | 19183-19272 |
| Mercurio | 05/19/81-05/20/81 | Bishop 4, 5, 7, 9, 10- Pipleline Rupture | 11391-11642 |
| | 09/14/81-09/15/81 | Doherty 29- Blockage of Intake Canal | 16593-16651 |
| | 09/15/81 | Board Question 8- Reactor Building Subsurface Subsurface Soil Mechanics | 16652-16680 |
| Meyer, J. | 08/17/81 | Doherty 30- Interconnection/ Grid Stability | 14583-14628 |

## WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Meyer, R. | 06/09/81-06/10/81 | Doherty 14, 25- Fuel Failure Detection Methods | 13616-13882 13975-13994 |
| | 06/10/81-06/11/81 | Doherty 3, 20(a), 39- Fuel Specific Enthalpy Gap Conductance and Cladding Swelling | 13995-14240 |
| Michaelson | 02/27/81 | Marrack 2(b)- Health Effects of High Voltage Transmission Lines | 7351-7492 |
| Minners | 12/07/81 | Doherty 8-ATWS | 20817-20898 |
| Mitchell | 10/06/81 | Doherty 38(b)-Cold Shutdown Within 24 Hours | 17887-17912 |
| Mokhtarian | 05/18/81 | Doherty 9- Containment Buckling | 11038-11140 |
| Montalbano | 06/02/81 | TexPirg 12- Cable Fires | 12541-12638 |
| | 08/18/81 | Doherty 10- Diesel Generator Reliability | 14698-14805 |
| Moon | 01/23/81 | Final Supplement to Final Environmental Statement | 3869-3871 |
| | 12/07/81 | Doherty 26-Stud Bolts | 20898-20938 |
| | 12/09/81 | Board Question 2- Use of WASH-1400, | 21125-21134 |
| | | Board Question 7- On-site transportation accident, | 21134-21153 |
| | | Board Questions- Un-Resolved Safety Issues | 21216-21293 |
| Murri | 09/18/81 | Schuessler Consolidated 1- Emergency Planning | 17401-17567 |

WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Nehemias | 06/01/81 | TexPirg AC 21-Occupational Exposure | 12370-12402 |
| Nuta | 11/17/81 | Doherty 5- Suppression Pool Uplift | 20237-20275 |
| Oprea | 10/07/81-10/08/81 | TexPirg AC 31-Technical Qualifications | 18053-18402 |
| Pappone | 06/01/81 | Board Question 4B-GDC-50 | 12486-12502 |
| Pearring | 09/15/81 | Doherty 29-Blockage of Intake Canal | 16688-16703 |
| Perl | 02/11/81-02/12/81 | Energy Alternatives | 5529-5974 |
| Petersen, T. | 10/30/81 | Bishop 6-LPG Pipeline | 19857-19862 |
| Peterson, D. | 10/27/81 | Doherty 28-Control Rod Ejection | 18993-19054 |
| Peterson, J. | 11/19/81 | Financial Qualifications | 20418-20606 |
| Ranganath | 06/05/81 | TexPirg AC 39 Generic All-Fracture Toughness | 13276-13328 |
| Ranzau | 08/25/81-08/26/81 | TexPirg AC 28-Post Accident Monitoring | 15739-16107 |
| Reed | 02/26/81 | Marrack 2(c)-Transmission Line Impacts on Water Fowl | 7174-7350 |
| | 05/12/81 | TexPirg AC 1,-Reactor Pressure Vessel Transportation | 10115-10157 |
| | 05/13/81 | Bishop 23(a),(c); Conn 2; Cummings 4; Doggett 2; Hinderstein 5; Johnson 5-2/6-2; Lemmer 2; TexPirg 1-Site Alternatives to ACNGS | 10225-10631 |

WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Read | 09/17/81 | Doherty 40-Part 100 Releases | 17332-17386 |
| Robertson | 08/21/81, 08/24/81 | Doherty 8- Board Question 3-ATWS | 15512-15642 |
| | 08/25/81- 08/26/81 | Board Question 4A- Combustible Gas Control, TexPirg AC 52- Outside Containment Sampling | 15739-16107 |
| Root | 06/05/81 | Doherty 50-Jet Pump Beams | 13267-13272 |
| Ross | 06/02/81 | Doherty 48-Control Rod Drive Return Line | 12684-12736 |
| Sanders | 01/21/81, 02/03/81, 02/04/81, 02/05/81- 02/06/81 | TexPirg 2 & 4, Griffith 4, McCorkle 2- Aquatic Ecology of Cooling Reservoir | 3243-3468 4244-4338 4499-4514 4701-5096 |
| | 05/12/81 | TexPirg AC 1, Reactor Pressure Vessel Transporation | 10115-10157 |
| | 05/13/81- 05/14/81 | Bishop 23(a), (c); Conn 2; Cumings 4; Doggett 2; Hinderstein 5; Johnson 5-2/6-2; Lemmer 2; TexPirg 1- Site Alternatives to Allens Creek | 10225-10631 |
| Schemel | 09/14/81 | TexPirg AC 28- Post Accident Monitoring | 16415-16593 |
| Schlicht | 01/16/81, 01/19/81, 01/20/81, 01/23/81 | TexPirg 2 & 4 McCorkle 2- Cooling Lake Recreational Benefits & Dam Extension | 2511-3221 3792-3840 |
| | 03/02/81- 03/03/81 | Marrack 2(c)-Transmission Lines/Waterfowl | 7503-7809 |
| Schoenberger | 03/05/81- 03/06/81 | Hinderstein 5- Coastal Site | 8570-8805 |

## WITNESS INDEX

## WITNESS INDEX

| Witness | Date | Testimony | Transcript Pages |
|---|---|---|---|
| Van Sickle | 03/04/81 | Hinderstein 5- Coastal Sites | 8039-8173 |
| Weingart | 06/01/81 | TexPirg AC 36, McCorkle 17- Charcoal Adsorber Fires | 12403-12460 |
| | 08/25/81 | Board Question 4A- Combustible Gas Control | 15739-15937 |
| | 11/18/81 | TexPirg AC 34- Hydrogen Monitoring | 20339-20378 |
| White | 03/16/81-03/17/81 | Bishop 1- Population Projections | 8885-9147 |
| Williams | 05/21/81-05/22/81 | Doherty 3- Fuel Specific Enthalpy, Doherty 39- Fuel Swelling, Doherty 20(a)- Gap Conductance | 11718-12228 |
| | 06/08/81 | Doherty 25- Fuel Failure/Flow Blockage, Doherty 14- Main Steam Line Radiation Monitor | 13364-13607 |
| Woodard | 05/19/81 | TexPirg 6, McCorkle 11- Aircraft Hazards | 11310-11351 |
| Woodson | 02/10/81 | TexPirg 5- Solid Waste Combustion | 5394-5473 |

DB05:N